ATOMIC ENERGY AUTHORITY
CAIRO, EGYPT

INTERNATIONAL ATOMIC
ENERGY AGENCY
VIENNA, AUSTERIA

REGIONAL (AFRA IV - 11), (RAF/4/O12)
TRAINING COURSE
ON
"RESEARCH REACTOR OPERATIONAL PERSONNEL"
(15 -26) MARCH, 1997
CAIRO, EGYPT
ORGANIZED
BY
IAEA - EAEA

COURSE DIRECTOR
PROF. M.K. SHAAT
AFRA PROJECT CO-ORDINATOR

SUPERVISED
BY
PROF. H. F.ALY
PRESIDENT OF EAEA
AFRA NATIONAL CO-ORDINATOR

Volume (1)

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B. Practical Part:

1- Visit to the ET-RR-1 Reactor systems and Experimental Facilities.
2- Operation Procedure of ET-RR-1
3- Carring out Experiments:
   - Criticality at different power Levels
   - Power Calibration
   - Control Rod Calibration
   - Temperature Coefficient of Reactivity
   - Neutron and gamma flux Mapping
   - Scram due to loss of Flow
   - Decay Heat Measurement by Data Aquisition System

IAEA - EGYPTIAN ATOMIC ENERGY AUTHORITY (EAEA)
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OURSE OPENTNG

9 : 00 - 9 : 30	Registration	
9 : 30 - 9 : 40	Prof. M.K.Shaat	(Course Director)
9 : 40 - 9 : 50	Prof. M.A. Gomaa	President of (EAEA) and National Co-ordinator
9 : 50 - 10: 00	Prof. A. H. Mariy	Head of Reactors Dept.
10:00 - 10 :10	Prof. M.A. Sultan	Reactors Dept.
10: 10 - 10:20	Mr. V, DIMIC	IAEA
10: 20 - 10:40	Break	
10: 40	Course Program	

C7-RAF-4.012-005

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(as of 1997.02.13)

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Week 2

TIME	SATURDAY 22 MARCH 97	SUNDAY 23 MARCH 97	MONDAY 24 MARCH 97	TUESDAY 25 MARCH 97	WEDNESDAY 26 MARCH 97
9:30 - 10:20	Reactor Safety H. Bock, IAEA	Incore Fuel Management M. Mekhael	Re furbishing and upgrading of Research reactors M. Khattab	Calibration of radiation detectors W. Aziz	Shielding Calculations M. Nagy
10:30 - 11:20	Reactor Management H. Bock, IAEA	Description of the ET-RR-1 reactor M. Khattab	Operation of ET-RR-1 (practical) S. Shibl	Water Chemistry M. Marawan	Safe guards requirements M. Sultan
11:20 - 11:40	COFFEE BREAK				
11:40 - 12:30	Computers in measurement and data processing M. Shaat	Visit to the ET-RR-1 reactor. A. El-kafas	Criticality, Power Calibration (practical) S. Sheibl	Temp. Coefficient measurement (practical) A. El-Kafas	
12:40 - 13:30	Description of experiments E. Saad	Visit to the ET-RR-1 Experimental Facilities A. El-kafas	Control rod Calibration (practical) S. Sheibl	Decay heat After ET-RR-1 shut down (practical) A. El-Kafas	Course Evaluation H.Bock, M. Sultan M. Shaat
13:40 - 14:30	Description of experiments W. Aziz	Neutron and gamma flux mapping H. Bock, IAEA	Control rod Calibration (practical) S. Sheibl	Decay heat measurement (practical) A. El-Kafas	----------
14:30	LUNCH				

REGIONAL (AFRA) TRAINING COURSE FOR REACTOR OPERATIONAL PERSONNEL
CAIRO - EGYPT
(15 - 26) MARCH 1997

Week 1

TIME	SATURDAY 15 MARCH 97	SUNDAY 16 MARCH 97	MONDAY 17 MARCH 97	TUESDAY 18 MARCH 97	WEDNESDAY 19 MARCH 97
9:30 - 10:20	Course Opening	Basic Nuclear Physics *M. Mekhael*	Heat Transfer and fluid flow *M. Khattab*	Radiation detection measurements (GM count. Sci.,) W. Aziz	Spent Fuel and Waste Management *R. El-Serogy*
10:30 - 11:20	Course objectives, review of the course programme *IAEA, DIMIC*	Reactor Physics *M. Mekhael*	Heat transfer and fluid flow *A. Abdalla*	Dosimetry (TLD, materials, read not systems) S. Yosef	Handling of Radioisotopes R. El-Shinawi
11:20 - 11:40	COFFEE BREAK				
11:40 - 12:30	Reports of participants	Reactor physics *M. Mekhael*	Reactor Instrumentation (detectors) *M. Shaat*	Radiation Protection at R.R. *M. Gomaa*	Review of Experiments for Research Reactor *A. Hassan*
12:40 - 13:30	Reports of participants	Reactor Kinetics *M. Shaat*	ET-RR-1 Instrumentation *M. Shaat*	Biological Effects of Ionizing Radiation A. El-Naggar	Silicon Doping *M. Sultan*
13:40 - 14:30	Reports of participants	Reactor Kinetics *M. Shaat*	ET-RR-1 Instrumentation *M. Shaat*	Reactor operation principles M. Shaat	Emergency plan *M. Gomaa*
14:[illegible]	LUNCH				

Lec. 1

SOME BASIC ASPECTS ON REACTOR THEORY

Prof. Dr.M.L.Michaiel
Reactors Department,Nuclear Research Centre,
Atomic Energy Anthority,Cairo,Egypt

ABSTRACT

This lecture deals with some basic subjects concerning reactor theory, namely:

- neutron multiplication,
- material and geometrical bucklings,
- reflected reactors and reflector saving, and
- reactor control including fuel deplation, fission products and temperature coefficients.

1. Neutron Multipllcation

A fission process is accompanied by the production of two fragments as well as 2-3 neutrons beside the libaration of energy. These neutrons produced by flssion having a large spectrum of energy may have any of the following reactions:

- Scattered by a nucleus of the fuel, moderator or constructing material and lose some (or all) of its energy. depending on the mass of this nucleus as well as the angle of scattering.
- Caputred by a nucleus of the elements inside the reactor core depending on the capture cross-section of this reaction.
- Absorbed by a fissile nucleus giving an exited compound nucleus that results in a fission into two fragments and neutrons and energy. This depends on the fission probability of this interaction (fission cross- section).
- The neutrons may leak out of the core directly (fast neutrons) or after one or several collisions (thermal neutrons) .

When the results of all these interactions is that one neutron (in a generation) gives one neutron in the next generation. a chain reaction is sustained. Taking as an example 1000 fast neutrons produced from fissions in one generation:

- Some neutrons are absorbed in fissile and fertile materials before being thermalized.
- Some of these absorbed fast neutrons make fission.
- The other neutrons which are escaped from absorption in the fast range of energy are scattered till they reach the thermal energy range.

- Some of these thermal neutrons are absorbed in the fuel while the others are absorbed in the moderator or constructing materials.
- Thermal neutrons absorbed in fuel may make fissions. -
- Neutron leakage in the fast range of energy as well as in thermal range must be considered.

The factor giving the ratio between neutrons escaped from fast (and resonance) absorption to all fast neutrons absorbed is called resonance escape probability and is denoted "P".

The ratio between neutrons production in the fast and thermal ranges to that in the thermal range only is called fast fission factor and is denoted "ε" .The factor relating thermal neutrons absorbed in fuel to those absorbed in fuel, moderator and constructing materials is called thermal utilisation factor "f". The ratio of neutrons produced per fission in thermal range to that absorbed in the fuel in the same range is called the regeneration factor "η"

These four factors can be defined in the following equations:

$$\eta = \left(\frac{\gamma \Sigma_f Ø}{\Sigma_a Ø}\right)_{fuel\ (thermal)}$$

$$f = \frac{(\Sigma_a Ø V)_{fuel}}{(\Sigma_a Ø V)_{fuel} + (\Sigma_a Ø V)_{mod.} + (\Sigma_a Ø V)_{c.m.}}$$

$$p = \frac{[(\Sigma_a ØV)_{fuel} + (\Sigma_a ØV)_{mod.} + (\Sigma_a ØV)_{c.m.}]_{thermal}}{[(\Sigma_a ØV)_{fuel} + (\Sigma_a ØV)_{mod.} + (\Sigma_a ØV)_{c.m.}]_{thermal} + (\Sigma_a ØV)_{fast}}$$

$$\varepsilon = \frac{(\gamma\Sigma_f \emptyset)_{thermal} + (\gamma\Sigma_f\emptyset)_{fast}}{(\gamma\,\Sigma_f\emptyset)_{thermal}}$$

Where : γ is the reproduction factor or the average number of neutrons produced per fission.

Σ_f macroscopic fission cross-section.

Σ_a macroscopic absorption cross-section.

V is the volume.

Ø is the average flux.

fuel: means in fuel.

mod.: means in moderator.

c.m.: means in constructing materials.

fast: means in fast range of energy.

thermal: means in thermal range of energy.

The product of these four factors gives what is called the infinite multiplication factor K_∞ . where the reactor is considered of infinite dimension. i.e., there is no neutron leakage.

$K_\infty = \eta\, f\, p\, \varepsilon$

$$= \frac{(\gamma\,\Sigma_f\,\emptyset)_{thermal} + (\gamma\Sigma_f\emptyset)_{fast}}{[(\Sigma_a\emptyset V)_{fuel} + (\Sigma_a\emptyset V)_{mod.} + (\Sigma_a\emptyset V)_{c.m}]_{thermal} + (\Sigma_a\emptyset V)_{fast}}$$

$$K_\infty = \frac{\text{total production}}{\text{total absorption}}$$

Since. practically there are some neutrons that leak out of the reactor core in the thermal and fast ranges of energy. this leakage must be considered in the multiplication factor calculaions:

the fast nonleakage probability $= e^{-B^2\tau}$

the thermal nonleakage probability $= \dfrac{1}{1 + L^2 B^2}$

where : B^2 is the material buckling (to be defined later).

τ is the neutron age.

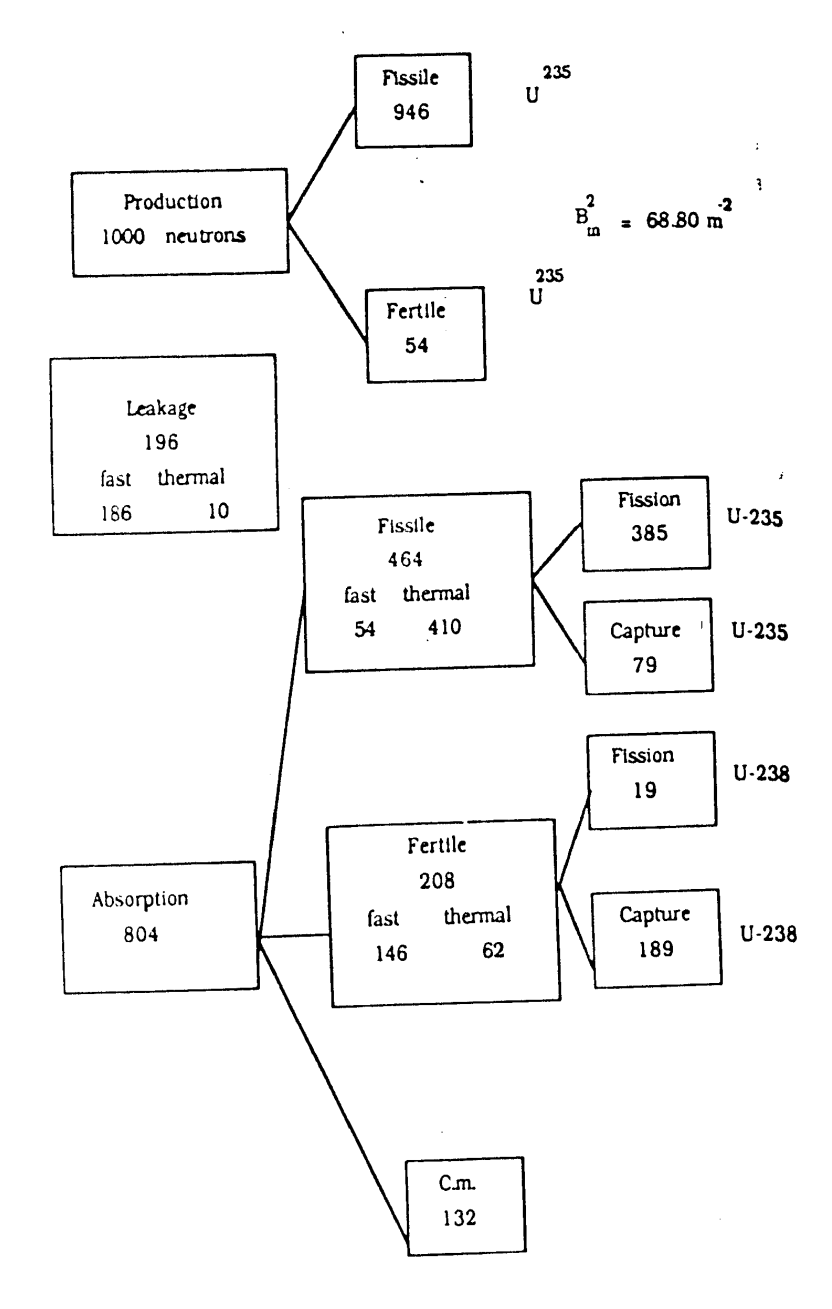
L^2 is the diffusion area.

Thus. the effective multiplication factor can be given as:

$$K_{eff} = \frac{K_\infty e^{-B^2\tau}}{1 + L^2 B^2} = \frac{K_\infty}{(1 + B^2 \tau)(1 + L^2 B^2)} = \frac{K_\infty}{(1 + M^2 B^2)}$$

$M^2 = L^2 + \tau$ is called the migration area.

As an example. the following table gives the neutron balance for a UO_2 light water reactor:



Fissile
946
U 235
Production
1000 neutrons
$B_m^2 = 68.80\ m^{-2}$
Fertile
54
U 235
Leakage
196
fast thermal
186 10
Fissile
464
fast thermal
54 410
Fission
385
U-235
Capture
79
U-235
Fission
19
U-238
Absorption
804
Fertile
208
fast thermal
146 62
Capture
189
U-238
C.m.
132

$\eta = 1.7984$

$f = 0.7993$

$p = 0.8184$

$\varepsilon = 1.0571$

$M^2 B^2 = 0.2434$

$K_\infty = 1.2434$

$K_{eff} = 1.0000$

C.m. absorption

$O_2 = 0$

$Cr = 14$

$Ni = 10$

$Fe = 44$

$H_2O = 64$

Material and geometrical bucklings:

The diffusion equation for one group of neutrons can be written in the form:

$$+ D \nabla^2 \emptyset - \Sigma_a \emptyset + \Sigma_a k \emptyset = o$$

where: D: diffusion coefficient.

Σ_a : macroscopic absorption cross-section.

K : multiplication factor.

$\emptyset$: flux.

This equation may take the wave equation form:

$$\nabla^2 \emptyset + B^2 \emptyset = o$$

where: $B^2 = \frac{(K-1)\Sigma_a}{D} = \frac{K-1}{L^2} \rightarrow \frac{K-1}{M^2}$ (in two groups of neutrons)

L^2 is the diffusion area $= \frac{D}{\Sigma_a}$

M^2 is the migration area $= L^2 + \tau$

τ is the age.

The quantity B^2 is given the name buckling since it represents ($\nabla^2 \emptyset/\emptyset$) a measure of the curveture (or buckling) of the neutron flux. B^2 may be calculated from:

(1) The wave equation as $B^2 = \nabla^2 \emptyset/\emptyset$. i.e., depending on the geometry of the system (plane. cylinder. sphere).

(2) The relation $B^2 = \frac{k-1}{M^2}$, i.e.. depending on the materials inside the system (reactor).

In the first case it is called geomterical buckling but in the second it is called material buckling.

N.B.: Since in the solution of the wave equation. there are several eigen-values correspond to the fundamental mode and the harmonics. only the fundamental (smallest eigen-value) will be considered.

In a critical system:
geometrical Buckling = material Buckling

$$B_g^2 = B_m^2$$

for $B_g^2 < B_m^2$ the critical radius is larger. and the reactor is super-critical. $K_{eff} > 1$

for $B_g^2 < B_m^2$ the critical radius is smaller. and the reactor is sub-critical. $K_{eff} < 1$

Geometrical Buckling:

As mensioned above the geometrical buckling depends on the shape of the reactor core (slab, sphere, cylinder, etc.). As an example for getting an expression for the geomterical buckling we will consider cylindrical reactor (most widely used).

Consider a cylindrical reactor of finite height, the laplacian operator is expressed in the wave equation as:

$$\frac{\partial^2 \emptyset}{\partial r^2} + \frac{\partial \emptyset}{r \partial r} + \frac{\partial^2 \emptyset}{\partial z^2} + B_g^2 \, \emptyset \, (r, z) = o$$

the Boundary conditions are:

$\emptyset$ (r,z) is finite and non-negative.

$\emptyset$ (r,z) = o at r = R_o or z = H

by separating the variables:

$\emptyset$ (r,z) = R (r). Z (z)

thus, $\frac{1}{R} \left(\frac{d^2 R}{dr^2} + \frac{1}{r}\frac{dR}{dr}\right) + \frac{1}{Z}\frac{d^2 Z}{dz^2} + B_g^2 = o$

$\therefore \frac{1}{R} \left(\frac{d^2 R}{dr^2} + \frac{1}{r}\frac{dr}{dr}\right) = -\alpha^2$

and. $\frac{1}{Z}\frac{d^2 Z}{dz^2} = \beta^2$

where : α^2 and β^2 are constants. $B_g^2 = \alpha^2 + \beta^2$

Putting X = α r and rearranging we get:

$$x^2 \frac{d^2R}{dx^2} + x \frac{dR}{dx} + x^2 R = o$$

which is the zero order Bessel equation.

(a) For positive values of α^2 , the general solution is

$$R = AJ_o (\alpha r) + CY_o (\alpha r)$$

where Jo and Yo are Bessel functions of the first and second kinds respectively of zero order.

(b) For negative values of α^2 the Bessel equation will be changed to be modified Bessel equation and the solution will be:

$$R = A' I_o (\alpha r) C'K_o (\alpha r)$$

For multiplying medium α is positive and since at

$r = o$, Y_o tends to $- \infty$, the coefficient $C \rightarrow o$, thus,

$$R = AJ_o (\alpha r)$$

at $r = R_o$, flux = o

$\therefore A J_o (\alpha R_o) = o$ $\qquad$ $\therefore \alpha R_o = 2.4048$

or $\alpha = (\frac{2.4048}{R_o})$ and $\alpha^2 = (\frac{2.4048}{R_o})^2$

also : $\frac{d^2 Z}{dz^2} + \beta^2 Z = o$

$$z = A_1 \cos (\beta z) + C_1 \sin (\beta z)$$

from symmetry $\quad C_1 = o$

$$Z = A_1 \cos (\beta z) = o \text{ at } z = \frac{1}{2} H$$

$$\beta \frac{H}{2} = \frac{\pi}{2} \text{ or } \beta^2 = (\frac{\pi}{H})^2$$

$$\therefore \quad B_g^2 = (\frac{2.4048}{R_o})^2 + (\frac{\pi}{H})^2$$

$$\Phi(r,z) = A''J_o\left(\frac{2.4048\, r}{R_o}\right)\cos\left(\frac{\pi z}{H}\right)$$

II. REFLECTED REACTORS

As discussed before, the multiplication factor depends on the neutron leakage from the reactor core. Thus, this factor increases by decreasing neutron leakage or in other words, the critical fuel used decreases by decreasing neutron leakage, i.e., saving of fuel depends on minimising leakage. This is done by surrounding the reactor core by material that can reflect back the neutrons. The reflector must have the following properties:

- Very small neutron absorption cross-section.
- It must be of light elements so as to reflect back the neutrons and decrease its energy to minimum in the same time, i.e., behaves as reflector and moderator.
- Small scattering mean free path to minimize the distance traveled by the neutron before being scattered. The most common moderator materials are H_2O, D_2O, graphite and Be.

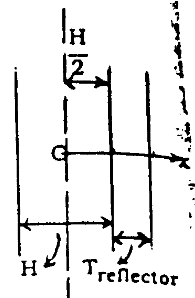
Reflector Savings:

As mensioned above, the presence of the reflector reduces the critical size of the fuel. The reflector savings (for an infinite plane slab) may be defined by:

$$\delta = \frac{1}{2} H_o - \frac{1}{2} H$$

where : H_o is the critical thickness of the bare slab.

H is the thickness with a reflector.



$$\therefore H_o = \frac{\pi}{B_c} \qquad . \ B_c \text{ Core buckling}$$

$$\delta = \frac{\pi}{2B_c} - \frac{1}{2} H \text{ or } \frac{1}{2} H = \frac{\pi}{2B_c} - \delta$$

The one group diffusion eqation may be written as:

$$D_o \nabla^2 \emptyset_c - \Sigma_{ac} \emptyset_c + k \Sigma_{ac} \emptyset_c = o \qquad \text{(core)}$$

$$D_r \nabla^2 \emptyset_r - \Sigma_{ar} \emptyset_r = o \qquad \text{(reflector)}$$

which can be written in the form:

$$\nabla^2 \emptyset_c + B_c^2 \ \emptyset_c = o$$

$$\nabla^2 \emptyset_r - k^2{}_r \emptyset_r = o$$

where : $B_c^2 = \frac{k-1}{L^2}$; $K_r^2 = \frac{-\Sigma_{ar}}{D_r}$

The solution of these equations (taking into consideration the positive x - direction and the flux is symmetrical, finite and non-nagative) will be:

$$\emptyset_c (x) = A \cos (B_c x)$$

$$\emptyset_c (x) = A' \cosh (k_r x) + C' \sinh (K_r x)$$

But since: $\varnothing_r (x = \frac{1}{2} H + T) = o$

$$\varnothing_r (x) = C \sinh K_r (\frac{1}{2} H + T - x)$$

From the boundary conditions:

$$\varnothing_c (\frac{1}{2} H) = \varnothing_r (\frac{1}{2} H)$$

$$D_c \nabla \varnothing_c (\frac{1}{2} H) = D_r \nabla \varnothing_r (\frac{1}{2} H)$$

$$A \cos (B_c \frac{H}{2}) = C \sinh (K_r T)$$

$$AD_c B_c \sin (B_c \frac{H}{2}) = C K_r D_r \cosh (K_r T)$$

by division

$$D_c B_c \tan (B_c \frac{H}{2}) = Dr\ Kr \coth (Kr\ T)$$

Substituting for $\frac{H}{2}$:

$$D_c B_c \tan (\frac{\pi}{2} - B_c \delta) = D_r K_r \coth (K_r T)$$

or ,

$$D_c B_c \cot (B_c \delta) = D_r K_r \coth (K_r T)$$

or,

$$\delta = \frac{1}{B_c} \left[\tan^{-1} \frac{D_c B_c}{D_r K_r} \tanh (K_r T) \right]$$

For H large B is small ($\tan^{-1} x = x$ for small x)

Thus. $\delta = \frac{D_c}{D_r K_r} \tanh (K_r T)$

If. $D_c \approx Dr$ $\therefore \delta = L_r \tanh (\frac{T}{L_r})$

III. REACTOR CONTROL

In the steady state operation, the effetive multiplication factor must remain unity. However, when the reactor operates for a long period, the multiplication factor will decrease due to several factors such as:

- The fuel is depleted.
- Some fission products have a poisoning effect (Σ_a is large).
- The temperature increases and consequently the cross-sections and density decrease resulting in a decrease in the multiplication factor for the majority of reactors (except for example some types of Pu reactors).

Due to these factors as well as other factors, the effective multiplication factor must exceed unity. The excess reactivity is controlled by several methods of control such as:

(1) Using control rods.
(2) Insertion or removal of fuel rods.
(3) Increasing or decreasing reflector thickness.
(4) Introducing burnable poisons.
(5) Introducing poisons in reflectors or moderators.
... . etc.

III. 1. Fuel Depletion

As the reactor operates, the uranium fuel is depleted but in the same time some fissile and fertile isotopes are formed thus:

$$_{92}U^{235} + {}_0n^1 \rightarrow {}_{92}U^{236}$$

$$_{92}U^{238} + {}_0n^1 \rightarrow {}_{92}U^{239} \xrightarrow{\beta} {}_{93}NP^{239} \xrightarrow{\beta} {}_{94}Pu^{239}$$

$$_{94}Pu^{239} + {}_0n^1 \rightarrow {}_{94}Pu^{240} \xrightarrow{+n} {}_{94}Pu^{241} \xrightarrow{+n} {}_{94}Pu^{242}$$

$$_{94}Pu^{241} \xrightarrow{\beta} {}_{95}Am^{241}$$

The rate of change of any of these isotopes can be given as:

$$\frac{dN_i(t)}{dt} = -\sigma_{ai}\, \emptyset N_i + \sigma_{ci-1}\, \emptyset N_{i-1} = a_i N_i + b_{i-1}\, N_{i-1}(t)$$

where : N_i is the number of nuclei/ C.C. of the isotope "i".

N_{i-1} is the number of nuclei/ C.C. of the isotope "i-1".

σ_{ai} is the microscopic absorption cross-section of the isotope "i".

σ_{ci-1} is the microscopic capture cross-section of the isotope "i-1").

The solution of this differential equation may be given in the form:

$$N_i = \sum_{l=1}^{i} \alpha_{il}\, e^{-a_l t} \qquad \alpha_{il} = \frac{b_{(i-1)}}{a_l - a_i}(\alpha_{i,l-1})$$

$$\alpha_{ii} = n_i - \sum_{l=1}^{i-1} \alpha_{il} \qquad l = 1, 2, \ldots, i-1$$

n_i = initial Concentration

The concentrations of U235 and U238 decreases by time of operation of the reactor, while that of U236, Pu240, Pu241 increases.

The plutonium isotopes increases by time till each of them reaches a saturation state.

III.2. Fission Products

Among, the different isotopes produced from fission there are some having very high absorption cross-sections which decreases the thermal utilization factor "f" and consequently the multiplication factor decreases. The most important of these isotopes are Xe^{135}, Sm^{149}, Gd^{155}, Gd^{157}, ... etc. Xe^{135} and Sm149 have beside the high absorption cross-sections a relatively high fission yields.

The rate of formation of any of these fission products may be given by the relation:

$$\frac{d_z B^A(t)}{dt} = \lambda_{z-1}^{A} C_{z-1}^{A}(t) + \Sigma_{f5}(t) \, \varnothing(t)_z \gamma^{A} +$$

$$\sigma_{az}^{A-1} \, \varnothing(t) \, {}_zB^{A-1}(t) - \lambda_z^{A} \, {}_zB^{A}(t) - {}_z\sigma_a^{A} \, \varnothing(t) \, {}_zB^{A}(t)$$

where: ${}_zB^A$ is the concentration of the element under consideration.

λ_{z-1}^{A} is the decay constant of the element of the same mass number of the isotope under consideration and atomic number less by one.

C_{z-1}^{A} is the concentration of the element of the same mass number and less by one of the atomic number from the isotope under consideration.

Σ_{f5} (t)is the macroscopic fission cross-section for U^{235}.

$\emptyset$ (t) is the average flux.

σ_{a}^{A-1} is the absorption cross-section of the isotope of mass number less by one and the same atomic number of the isotope under consideration.

B_{z}^{A-1} is the isotope of the same atomic number but of mass number less by one of the isotope under consideration.

λ_{z}^{A} is the decay constant of isotopes under consideration.

σ_{a}^{A} is the absorption cross-section of isotope under consideration.

σ_{z}^{A} fission yield of the isotopes under consideration.

The solutions of this system of equations are:

$$
{}_{z}B^{A}(t_i) = \left[\lambda_{z-1}^{A} \frac{{}_{z-1}C^{A}(t_{i-1}) + {}_{z-1}C^{A}(t_i)}{2} \right.
$$

$$
+ {}_{za}\sigma^{A-1} \emptyset \frac{{}_{z}B^{A-1}(T_{i-1}) + {}_{z}B^{A-1}(t_i)}{2}
$$

$$
\left. + \Sigma_{f5} \emptyset \, {}_{z}\gamma^{A} \cdot \left[\frac{1 - e^{-({}_{z}\sigma^{A}\phi + {}_{z}\lambda^{A})t}}{(\lambda_{z}^{A} + {}_{z}\sigma_{a}^{A}\emptyset} \right] \right]
$$

From this relation it is obvious that the concentration depends on the neutron flux "Ø". If we define the poisoning P as:

$$P = \frac{\Sigma_p}{\Sigma_u}$$

Σ_p = macroscopic absorption cross-section of the fission products.

Σ_u = macroscopic absorption cross-section of the uranium.

The decrease in reactivity (Keff-1) due to the presence of the poison will be:

$$\rho = \frac{K_{eff} \; K'_{eff}}{K_{eff}} = - fP$$

(f is the thermal utilization factor before the presence of poison).

III.3. Temperature Coefficients

The release of energy from fission causes an increase of the temperature in the core. Such an increase will affect the reactivity for at least two reasons.:

- The mean energy of thermal neutrons increases and hence their absorption will be affected because the nuclear cross-sections vary with energy. (nuclear temperature coefficients).
- The mean free paths and the nonleakage probabilities are functions of the density, which changes with temperature (density temperature coefficients).

Since: in the two group analysis: (for large reactors)

$$K_{eff} = \frac{K_\infty}{(1 + L^2 B^2)(1 + \tau B^2)} = \frac{\eta P \varepsilon f}{(1 + L^2 B^2)(1 + \tau B^2)} = \frac{\eta p \varepsilon f}{(1 + M^2 B^2)}$$

$$\therefore \frac{\partial K_{eff}}{K_{eff}\partial T} = \frac{1}{\eta}\frac{\partial \eta}{\partial T} + \frac{1}{\varepsilon}\frac{\partial \varepsilon}{\partial T} + \frac{1}{P}\frac{\partial P}{\partial T} + \frac{1}{f}\frac{\partial f}{\partial T}$$

$$- \frac{(K_{\infty} - 1)}{K_{\infty} M^2}\frac{\partial M^2}{\partial T} - \frac{K_{\infty} - 1}{K_{\infty} B^2}\frac{\partial B^2}{\partial T}$$

where T is the absolute temperature.

Nuclear temperature coefficient:

Let us take η as an example : $\eta = \frac{\gamma \Sigma_f}{\Sigma_a} = \gamma \frac{\sigma_f}{\sigma_a}$

Thus, only the nuclear coefficient (variation of cross-sections) is considered. If the absorption cross-sections of fuel and moderator are considered to follow the 1/v law in the thermal energy range,

$$\sigma_a = \sigma_{ao} \left(\frac{T_o}{T}\right)^{\frac{1}{2}}$$

so, as the temperature increases the cross-section decreases.

Density temperature coeficcient:

Here, the diffusion area can be taken as an example:

$$L^2 = \frac{D}{\Sigma_a} = \frac{1}{3\,\Sigma_{tr}\,\Sigma_a}$$

But, since Σ_{tr} is most probably due to presence of the moderator and Σ_a is largely due to absorption of the fuel.

$$\text{where } L^2 = \frac{1}{3N_m\, N_f\, \sigma_{tr_m}\, \sigma_{af}}$$

where: m referes to moderator.

f referes to fuel.

N is the number of nuclei / C.C. (depend on the density).

The variation of L^2 with temperature is due mainly to:

- Nuclear change of σ_{tr_m} and σ_{af}
- Physical (density) change in N_m and N_f and particularly that of the moderator (N_m).

Leg. 2.

I. NUCLEAR REACTION

Prof. Dr. M. L. Michaiel

When two nuclei or a nucleus and a nucleon interact to produce two or more nuclear particles or gamma radiation, a nuclear reaction is said to have taken place. If the initial nuclei are denoted by a and b, and the product nuclei are denoted by c and d, the reaction can be represented by the equation

$$a + b \longrightarrow c + d$$

In the actual experimental arrangement the nucleus, say a is at rest in some sort of target, and the particle b is projected towards the target and the reaction is often written in an abbreviated form a (b,d) c.

For example, when oxygen is bombarded by energetic neutrons, one of the reactions that can occur is

$$O^{16} + n \longrightarrow N^{16} + H \;, \quad \text{or} \quad O^{16}(n,p)N^{16}$$

It is important to note the four fundamental laws governing the nuclear reaction:

(1) Conservation of nuclear mass; the number of nucleons before and after a reaction must be equal.

(2) Conservation of charge; the sum of the charges on all the particles before and after a reaction must be equal.

(3) Conservation of linear and angular momentum; the total momentum of the interacting particles before and after a reaction are the same since no external forces act upon the particles.

(4) Conservation of energy, total energy is conserved in all nuclear reactions. The total energy is the sum of the kinetic energy and the rest mass energy of the interacting particles.

II. INTERACTION NEUTRONS WITH MATTER

The operation of a nuclear reactor depends fundamentally on the way in which neutrons interact with atomic nuclei. It is necessary, therefore, to consider the nature of these interactions in some detail.

Neutrons interact with nuclei in a variety of ways. For instance, if the nucleus is unchanged in either isotopic composition or internal energy after interacting with a neutron, the process is called elastic scattering. On the other hand, if the nucleus, still unchanged in composition, is left in an excited state, the process is called inelastic scattering. The symbols (n,n) and (n,n') are often used to denote these processes. In refering to these interactions it is common to say that the incident neutron has been "scattered", elastically or inelastically, as the case may be, because a neutron reappears after the interaction. However, this term is somewhat misleading, since the emerging neutron may not be the same neutron that originally struck the nucleus.

Neutrons disappear in a reactor as the result of absorption reactions, the most important of which is the (n,γ) reaction. This process is also known as radiative capture, since one of the products of the reaction is γ-radiation. Neutrons also disappear in charged-particle reactions such as the (n,p) or (n,α) reactions. Occasionally, two or more neutrons are emitted when a nucleus is

struck by a high-energy neutron. The processes involved here are of the (n,2n) or (n,3n) type. A closely related process is the (n,pn) reaction, which also occurs with highly energetic incident neutrons. Finally, when a neutron collides with certain heavy nuclei, the nucleus splits into two large fragments with the release of considerable energy. This, of course, is the fission process.

In one way or another, most of these interactions must be taken into account in the design of a nuclear reactor. Before considering the specific interactions, however, it is necessary to set up a framework with which these interactions can be discussed quantitively.

II.1. Cross-sections

The interactions of neutrons with matter are described in terms of quantities known as cross-sections which are defined in the followingway. Consider a thin target of area a and thickness X-containing N atoms per unit volume, placed in a uniform, monodirectional beam of neutrons of intensity I, which strikes the entire target normal to its surface as shown in Fig. (2.1). In such an experiment, it is found that the rate at which interactions occur within the target is proportional to the beam intensity and to the atom density area, and thickness of the target. If for instance the area of target is doubled, the interaction rate is also doubled, provided, of course, the beam still has the same intensity over the entire target. On the other hand, if the

intensity is doubled, the number of interactions which take place in a given time is also doubled, and so on.

These observations can be summarized by the equation:

$$\text{Interaction rate (in the entire target)} = \sigma I \, Na \, X \tag{2.1}$$

where the proportionality constant, σ, is known as the cross-section. Solving Eq. (2.1) for σ gives

$$\sigma = \text{Interaction rate} \, / \, I \, Na \, X \tag{2.2}$$

However, Na X is equal to the total number of atoms in the target, and it follows therefore that σ is the interaction rate per atom in the target per unit intensity of the incident beam.

It may be noted that in view of the definition of beam intensity, I.a neutrons strike the target per second, and according to Eq. (2.1), $\sigma I \, Na \, X$ interact. The relative probability that any one neutron in the beam interacts is therefore,

$$\frac{\sigma I \, Na \, X}{Ia} = \left(\frac{\sigma}{a}\right)(Na \, X) \tag{2.3}$$

Since the quantity Na X is the number of nuclei in the target, it follows that σ/a is the probability per target nucleus that a neutron in that portion of the beam striking the target will interact. Therefore, since the area of the target is fixed by the experiment, the probability of an interaction is determined by σ alone. It is in this sense,

that of a probability of interaction, that the concept of cross-section has its widest application.

It was assumed in the preceding discussion that the incident neutron beam strikes the entire target, which is the case when the target is smaller in area than the beam. In many experiments the reverse is true, and the beam is smaller than the target. When this is the case, it is merely necessary to replace the target area appearing in the above formulae by the interaction area, that is, the area of the target exposed to the beam. The definition of cross-section, of course, is the same in both cases.

It can be seen from Eq. (2.2) that σ has the dimensions of area. Cross-sections are usually measured in units of barns, where 1 barn, abbreviated as b, is equal to 10^{-24} cm^2.

As mentioned in the introduction to this chapter, neutrons interact with nuclei in a number of ways, and it is convenient to describe each type of interaction in terms of a characteristic cross-section. Thus elastic scattering is described by the elastic scattering cross-section, σ_s: inelastic scattering by the inelastic scattering cross-section, σ_{si}: the (n,γ) reaction (radiative capture) by the capture cross-section, σ_c: fission by the fission cross-section, σ_f: etc. The sum of the cross-sections for all possible interactions is known as the total cross-section and is denoted by the symbol σ_t; that is

$$\sigma_t = \sigma_s + \sigma_{si} + \sigma_c + \sigma_f + \ldots \qquad (2.4)$$

The total cross-section measures the probability that in interaction of any type will occur when neutrons strike a target.

The sum of the cross-sections of all absorption reactions is known as the absorption cross-section and is denoted by σ_a. Thus,

$$\sigma_a = \sigma_c + \sigma_f + \sigma_p + \sigma_\alpha \qquad (2.5)$$

where σ_p and σ_α are the cross-sections for the (n,p) and (n,α) reactions. As indicated in Eq. (2.5) fission, by convention, is treated as an absorption process.

Finally, the difference between the total and elastic cross-sections is known as the nonelastic cross-section, and is usually denoted by σ_{ne}. In symbols, this is

$$\sigma_{ne} = \sigma_t - \sigma_s \qquad (2.6)$$

The nonelastic cross-section is occasionally called the "inelastic cross-section". This terminology is not correct, however, unless inelastic scattering is the only nonelastic process that can occur.

II.2. Neutron Interactions and Macroscopic Cross-sections

Suppose that a target of thickness X is placed in a monodirectional beam of intensity I_0 and that a neutron detector is place at some distance behind the target as

shown in Fig. (2.2). It will be assumed that both the target and the detector are small so that the detector subtends a small solid angle at the garget. In this case every neutron that interacts in the target will be lost from the beam, and only those neutrons that do not interact will enter the detector.

Let I(x) be the intensity of the noninteracted neutrons after penetrating the distance x into the target. In traversing an additional distance dx, the intensity of the beam will be decreased by the number of neutrons that have interacted in the thin sheet of thickness dx. In view of Eq. (2.1), this decrease in intensity is given by

$$-dI(x) = N\sigma_t I(x)dx \tag{2.7}$$

where N is the atom density of the target. The total cross-section must be used in Eq. (2.7) since by definition any interaction removes a neutron from the noninteracted beam. Equation (2.7) can be integrated with the result

$$I(x) = I_o \, e^{-N\sigma_t x} \tag{2.8}$$

The intensity of the noninteracted beam thus decreases exponentially with distance inside the target. The intensity of the noninteracted beam emerging from the target is then,

$$I(x) = I_o \, e^{-N\sigma_t X} \tag{2.9}$$

and this the intensity measured by the detector.

The product of the atom density and a cross-section, which appears in the exponential in Eq. (2.9), occurs quite frequently in the equations of reactor theory; it is given the special symbol Σ, and is called the macroscopic cross-section, $N\sigma_t = \Sigma_t$ is called the macroscopic total cross-section, etc. Macroscopic cross-sections evidently have the dimensions of cm^{-1}.

In terms of Σ_t, Eq. (2.7) can be written as

$$- dI(x) = \Sigma_t I(x)dx \tag{2.10}$$

Dividing this expression by I(x) gives

$$- \frac{dI(x)}{I(x)} = \Sigma_t dx \tag{2.11}$$

The quantity $dI(x)/I(x)$ in this equation is equal to the fraction of the neutrons that have penetrated the distance x into the target without interacting, which subsequently interact in the distance dx. This, in turn, is equivalent to the probability that a neutron which survives up to x interacts in the next dx. Thus, from Eq. (2.11), $\Sigma_t\, dx$ is the probability that a neutron interacts in dx, and it follows that Σ_t is the probability per unit path length that a neutron will undergo some sort of interaction. In a similar manner, it is easy to show that Σ_s the macroscopic scattering cross-section, is equal to the probability per unit path length that a neutron will undergo elastic scattering. Analogous interpretations hold for all other macroscopic cross-sections.

Returning to Eq. (2.8), it should be noted that in view of the fact that I(x) refers to those neutrons that have not interacted in penetrating the distance x, the ratio $I(x)/I = e^{-N\sigma_t x} = e^{-\Sigma_t x}$ is equal to the probability that a neutron can move through this distance without interacting. Now let the quantity p(x)dx be the probability that a neutron will have its first interaction in dx in the neighborhood of x. This is evidently equal to the probability that the neutron survives up to x without interaction times the probability that it does in fact interact in the additional distance dx. Since Σ_t is the probability of interaction per path length. p(x)dx is given by

$$\begin{aligned} p(x)dx &= e^{-\Sigma_t x} \cdot \Sigma_t \, dx \\ &= \Sigma_t \, e^{-\Sigma_t x} \, dx \end{aligned} \tag{2.12}$$

The first interaction probability distribution function p(x) can be used in a number of ways. For instance, the probability p(a,b) that a neutron will have its first interaction between x = a and x = b is simply the integral of p(x)dx between these limits. That is,

$$\begin{aligned} p(a,b) &= \Sigma_a \int_a^b e^{-\Sigma_t x} \, dx \\ &= e^{-\Sigma_t a} - e^{-\Sigma_t b} \end{aligned} \tag{2.13}$$

In particular, the probability that a neutron will interact at least once in an infinite medium is obtained by placing a = 0 and b = ∞ ; thus

$$p(o,\infty) = \int_0^\infty p(x)dx = 1$$

as would be expected.

The distance that a neutron moves between interactions is called a free path, and the average distance between interactions is known as the mean free path. This quantity, which is usually designated by the symbol λ (not to be confused with the radioactive decay constant or neutron wavelength), is equal to the average value of x, the distance traversed by a neutron without interaction, over the interaction probability distribution p(x), that is,

$$\begin{aligned} \lambda &= \int_0^\infty x\, p(x)\, dx \\ &= \Sigma_t \int_0^\infty x\, e^{-\Sigma_t x}\, dx \\ &= 1/\Sigma_t \end{aligned} \qquad (2.14)$$

II.3. Cross-sections of Mixtures and Molecules

Consider a homogenous mixture of two nuclear species X and Y, containing N_X and N_Y atoms/cm of each type, and let σ_X and σ_Y be the cross-sections of the two nuclei for some particular interaction. According to the discussion in the previous section, the probability per unit path that a neutron interacts with a nucleus of the first type is $\Sigma_X = N_X \sigma_X$ and with the second is $\Sigma_Y = N_Y \sigma_Y$. The total probability per unit path that a neutron interacts with either nucleus is, therefore,

$$\Sigma = \Sigma_X + \Sigma_Y = N_X \sigma_X + N_Y \sigma_Y \qquad (2.15)$$

If the nuclei are in atoms that the bound together in a molecule. Eq. (2.15) can be used to define an equivalent cross-section for the molecule. This is done simply by dividing the macroscopic cross-section of the mixture by the number of molecules per unit volume. If, for instance, the molecular formula is X_mY_n the resulting cross-section for the molecule is

$$\sigma = m\ \sigma_X + n\ \sigma_Y \qquad (2.16)$$

Equations (2.15) and (2.16) are based on the assumption that the nuclei X and Y act independently of one another when they interact with neutrons. In some cases, particularly for low-energy elastic scattering by molecules and solids, this assumption is not valid and Eqs. (2.15) and (2.16) do not apply.

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If the nuclei are in atoms that the bound together in a molecule. Eq. (2.15) can be used to define an equivalent cross-section for the molecule. This is done simply by dividing the macroscopic cross-section of the mixture by the number of molecules per unit volume. If, for instance, the molecular formula is X_mY_n the resulting cross-section for the molecule is

$$\sigma = m\,\sigma_X + n\,\sigma_Y \tag{2.16}$$

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Lag. 3

NUCLEAR REACTOR KINETICS

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ABSTRCT

For a nuclear reactor to operate at a constant power level, the rate of neutron production should be balanced by neutron loss via absorption and leakage.

Any deviation from this balance condition will result in a time-dependence of power level of the reactor.The items and equations describing the time-dependent behavior of nuclear reactors will be studied and analyzed.Also,the reactor transfer function will be evaluated.

1. INTRODUCTION :

The study of the behavior of the neutron population in a noncritical reactor is called *reactor kinetics*.

The degree of criticality of a reactor is usually regulated by the use of *control rods* or *chemical chim*. Contrtol rods are pieces or assemblies of neutron absorbing material whose motions have an effect on the multiplication factor of the system.Thus if a control rod is withdrawn from a critical reactor,the reactor tends to become supercritical;if a rod is inserted,the system falls subcritical.With chemical shim,control is accomplished by varying the concentration of a neutron-absorbing chemical usually boric acid(H_3BO_3),in a water moderator or coolant.

A reactor will become supercritical or subcritical if its properties are changed in a such way that its multiplication factor becomes different from unity.These changes are:

1)Control rod motion.

2)Fuel burnup.

3)Isotope production.

4)Temperature changes.

It should be noted that a reactor is initially fueled with more than the minimum amount of fuel necessary for criticality-if this were not the case and it were fueled

with only its minimum critical mass, the reactor would fall subcritical after the first fission.

Control rods or boric acid are introduced into the core to compensate for the excess fuel. Then as fissions occur and the fuel is consumed, the rods are slowly withdrawn or the H_3BO_3 concentration is reduced to keep the reactor critical.

Before going further we have to define some concepts before introducing the reactivity.

Multiplication factor: the multiplication factor k may be defined as the ratio of the number of neutrons in any one generation to the number of corresponding neutrons of the immediately preceding generation. If $k > 1$ the reactor will be supercritical. If $k = 1$ the reactor will be critical. If $k < 1$ the reactor will be subcritical.

The more common multiplication factor which is used in conjunction with specific reactors is k_{eff} (k effective), which is the effective multiplication factor for a given finite-sized reactor. *reactivity* is defined for a finite specific reactor in a similar manner to k_{ex} (k excess) as

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \qquad (1)$$

In this project the symbol δk will be used for reactivity, indicative of the amount the multiplication

factor of a specific reactor differs from unity or

$$\delta k = \rho = \frac{k_{eff} - 1}{k_{eff}} \qquad (2)$$

2. PROMPT AND DELAYED NEUTRONS :

Most of the neutrons released in fission(usually more than 99 percent)are emitted essentially at the instant of fission.These are called *prompt neutrons* distinguish them from the *delayed neutrons* that are released comparatively long after the fission event.

If most of the fissions result from the capture of neutrons which have been slowed down to thermal energies by collisions with the moderating material,the so-called thermal neutrons,the system is referred to as a thermal reactor.When most of the fission processes are caused by the absorption of neutrons of higher energy,sometimes called intermediate neutrons,the term intermediate reactor is used. If the main source of fissions is the capture of fast neutrons directly by the fuel without the neutrons having suffered appreciable energy losses,the system is called a fast reactor.

The time behavior of a reactor depends upon the various

properties of these two types of neutrons. The prompt neutons will be cosiderd first.

Prompt Neutron Lifetime :

Following their emission, the prompt fission neutrons slow down as the result of elastic and inelastic collisions with nuclei in the system.In thermal reactors,however,most of them succed in reaching thermal energies without being absorbed or escaping from the system.As thermal neutrons,they diffuse about in the reactor;some eventually are absorbed and some leak out.

The average time between the emission of the prompt neutrons and their absorption in a reactor is called the prompt neutron lifetime and is denoted by l^* .

If we consider that the value of l^* in infinite thermal reactor.it can be shown theoretically,and it has been found experimentally,that the time required for a neutron to slow down to thermal energies is small compared to the time that the neutron spends as a thermal neutron before it is finally absorbed.The average lifetime of a thermal neutron in an infinite system is called the mean diffusion time and is given the symbol t_d.It follows therefore,that

$$l^* = t_d \qquad (3)$$

also $$t_a = t_{dm}(1-f) \quad (4)$$

for example the prompt neutron life time in an infinite , critical thermal reactor consisting of a homogeneous mixture of U and unit density h2O at room temperature.

$$l^* = t_d = 2.1*10^{-4}(1-0.484) = 1.08*10^{-4} \text{ sec.}$$

3. Infinite Reactor with No Delayed Neutrons :

As noted earlier, the delayed neutrons play an important role in reactor kinetics. This is a remarkable fact since so very few fission neutrons are delayed-less than one percent for thermal fission in U^{235}. To understand the importance of the delayed neutrons, it is helpful to consider first the kinetics of a reactor in the absence of delayed neutrons that is, assuming that all neutrons are emittted promptly in fission.

Since the eventual absorption of a fission neutron begins a new generation of fission neutrons, it is clear that in the absence of delayed neutrons, l^* is also equal to the time between successive generations of neutrons in the chain reaction. This time is called the *mean generation time*.

$$N_F(t) = N_F(0)\exp\left(\frac{K\omega - 1}{l^*}\right) \quad (5)$$

$$N_F(t) = N_F(0)e^{t\backslash T} \quad (6)$$

Where $N(t)$ is the number of fissions occurring per

cm^3/sec at the time t.

so
$$T = \frac{l^*}{k_\infty - 1} \qquad ([illegible])$$

is called the reactor period-in the absence of delayed neutrons. The period of the reactor is that amount of time in which the reactor would take to change its level by a factor e=2.716.That is,when the reactor is in operation at a fixed power level,the period is infinite.Only when the reactor is changing its level is there a finite measurable period.

Suppose that the reactor described in the preceding example is critical up to time t =0,and then k_∞ is increased from 1.000 to 1.001.The period of this reactor is

$$T = \frac{10^{-4}}{1.001-1.000} = 0.1 \text{ sec}$$

the period computed is very short.Thus with a period of 0.1 sec,the reactor would pass through 10 periods in only 1 sec,and the fission rate(and power)would increase by a factor of e =22,000.Had the reactor originally been operating at a power of 1 megawatt,the system would reach a power of 22,000 megawatts in 1 sec,if it did not destroy itself first-as it undoubtedly would.

4. Mean Generation time with Delayed Neutrons :-

When the delayed neutrons are included,the generation time

is increased considerably and is equal to the sum of the prompt and delayed neutron lifetimes weighted by their relative yields.

$$l \simeq l_p + \sum_{i=1}^{6} \beta_i \bar{t}_i \qquad (8)$$

where β_i is the fraction of the fission neutrons which appear in the ith group.

t_i is the mean-life of a neutron in the ith group.

Return now to the last example which discussed earlier.by using table 2-2 for U^{235},the term $\beta_i \bar{t}_i$ in Eq.(8) is found to be 0.085 sec $\simeq$ 0.1 sec.Again $l^* \simeq 10^{-4}$ and from Eq.(8) l is about 0.1 sec,rather than 10^{-4} sec as in the last example.A 0.1 percent change in k_∞ according to Eq.(2-7) now leads to a period of T=0.1/0.001 =100 sec.With this period it would take 100 sec for the reactor power to increase by a factor of e and the system could easily be controlled by the motion of control rods.

5. Infinite Reactor with Delayed Neutrons :

Unfortunately,the precise effect of the delayed neutrons on the response of a reactor to a change in its multiplication factor cannot be found by simply replacing the mean generation time in the absence of delayed neutrons by the more appropriate expression given in Eq.(8) Instead

it is necessary to consider in detail the production and subsequent decay of each of the delayed neutron precursors in the reactor.

The reactivity equation.

Since the flux independent of position in an infinite reactor, the equation of continuity for the thermal neutrons

is
$$-\phi_T(t) + \frac{q_T(t)}{\Sigma_a} = t_d \frac{d\phi_T(t)}{dt} \qquad (9)$$

the $q_T(t)$, the thermal slowing-down density at the time t, has been used for the thermal source term. The function $q_T(t)$ has two parts : one from the slowing down of the prompt neutrons which is equal to $(1-\beta)k_\infty \Sigma_a \phi_T(t)$.

The other one from the slowing down of the delayed neutrons which equal to $p\sum_i \lambda_i C_i(t)$.

where $C_i(t)$ is the concentration at time t in atoms/cm of the precursors of the ith group, and λ_i is their decay constant. Then the total slowing down density from both prompt and delayed neutrons is therefore

$$q_T(t) = (1-\beta)k_\infty \Sigma_a \phi_T(t) + p\sum_i \lambda_i C_i(t) \qquad (10)$$

Inserting this result into Eq.(2-9) gives

$$[(1-\beta)k_\infty - 1]\phi_T(t) + \frac{p}{\Sigma_a} \sum_i \lambda_i C_i(t) = t_d \frac{d\phi_T(t)}{dt} \qquad (11)$$

The precursors decay at the rate $\lambda_i C_i(t)$ and their

concentration at any time is determined by the equation

$$\frac{dC_i(t)}{dt} = \beta_i \frac{k_\infty}{p} \bar{\Sigma}_a \phi_T(t) - \lambda_i C_i(t) \qquad (12)$$

by assuming the solutions of the form

$$\phi_T(t) = A\, e^{\omega t} \qquad (13)$$

$$C_i(t) = C_i e^{\omega t} \qquad (14)$$

where A, C, and ω are constants to be determined. Becauce Eqs. (11) and (12) are linear, the sum of all possible solutions of this form will provide the general solution to the problem. then by inserting Eq. (2-13) and (2-14) into Eq. (12) and solving for C_i gives

$$C_i = \frac{\beta_i k_\infty \Sigma_a A}{p(\omega + \lambda_i)} \qquad (15)$$

When this result is substituted into Eq. (2-11), the constant, A, cancels from the equation and the following is obtained :

$$(1-\beta) k_\infty - 1 + k_\infty \sum_i \frac{\lambda_i \beta_i}{\omega + \lambda_i} = \omega\, t_d \qquad (16)$$

Finally, if the constant β is replaced by $\sum_i \beta_i$ and the equation is rearranged, it can be put in the following form which is convenient for computations :

$$\frac{k_\infty - 1}{k_\infty} = \frac{\omega\, t_d}{1+\omega\, t_d} + \frac{\omega}{1+\omega\, t_d} \sum_i \frac{\beta_i}{\omega + \lambda_i} \qquad (17)$$

The quantity on the left-hand side of this equation is known as the reactivity of the infinite reactor and is given

the symbol ρ :thus

$$\rho = \frac{k_\infty - 1}{k_\infty} \qquad (18)$$

so equation (17) now becomes

$$\rho = \frac{\omega t_d}{1+\omega t_d} + \frac{\omega}{1+\omega t_d} \sum_i \frac{\beta_i}{\omega + \lambda_i} \qquad (19)$$

This equation is known as the *reactivity equation* and ocassionallly as the *inhour equation.*

When the fractions in Eq.(2-19) are cleared,a seventh degree polynomial in ω is obtained which has seven roots.Thus there are seven values of ω.The nature of the roots of Eq.(19) may be seen by the plotting the (RHS) of Eq.(19) as a function of ω as shown in Fig. 2-1.It will be observed that the RHS = 0 for ω = 0 and increases monotonically to unity with increasingly positive values of ω.When ω is negative,however, the RHS is singular for the six values of $\omega = -\lambda$ and at $\omega = -1/t$ .Also,in the limit as $\omega \rightarrow -\omega$,RHS $\rightarrow$ 1.

Reactivity Dollars: We have become aware of two methods of defining reactivity:the inhour and the percentage change in multiplication factor δk.The inhour is defined in terms of period above criticality and is a unit generally used for large reactors with small available excess reactivities.δk,or more exactly $\delta k/k$,is a single fiducial

unit expressing relative or percentage departure from criticality.Another unit,the dollar = 100 cents,is sometimes used as a double fiducial unit of reactivity.The dollar is defined in terms of the interval between delayed critical and prompt critical as

$$\$ = \frac{\delta k}{\gamma \beta} \qquad (20)$$

where γ is the effectiveness of delayed neutrons in producing fission compared with prompt neutrons.γ =1.05 for U^{235} and may gain be neglected for many engineering callculations.

6. Neutron Level.

The excess of neutrons in a finite reactor from one generation over the preceding generation per unit of those in the generation is then δk.If ther are initially n neutrons per cubic centimeter present in the core,insrease in each generation is $n\delta k$.If l is the effective time between succeeding generations,

$$\frac{dn}{dt} = \frac{\delta k}{l^{*}} n \qquad (21)$$

and integrating this equation yields

$$n = n_o e^{(\delta k / l^{*})t} \qquad (22)$$

where n_o is the number of neutrons per cubic centimeter initially and n is the number after a lapse of time t.

The number of neutrons in the core is proportional to the number of fissiona ocurring,and for 3*10 fissions per second 1 watt of power is produced.The power output of a reactor then is proportional to the number of neutrons in the core at any given instant.

7. Point Reactor Kinetics Eqn.

In the situation of delayed neutrons when present.Our neutron level equation becomes of the form

$$\frac{dn}{dt} = \frac{\delta k}{l^*} n - \frac{\beta}{l^*} n + \sum_i \lambda_i C_i \qquad (23)$$

where C_i is the concentration of the delayed neutrons emitted of group i . C_i is defined by

$$\frac{dC_i}{dt} = \frac{\beta_i}{l^*} n - \lambda_i C_i \qquad (24)$$

The rate of change of n has the contribution of the delayed neutrons subtracted from the prompt neutrons,but of course the concentration of delayed neutrons coming in from the past must be added to make up the total rate of change.

8. Solution of P.K.E. for Step-function Input.

The nature of the solutions of Eqs.(23)and (24) for step-function inputs is is known as follow.These kinetic equations can be combined to form a single differential equation of the seventh order in n.For astep input in δk the

solution will take the form

$$n(t) = n_0 \sum_{j=1}^{7} A_j e^{p_j t} \qquad (25)$$

where the first exponent P_1 has the same sign as δk, the input disturbance, and where the other six exponents are negative. It is evident that in the transformed form

$$n(s) = n_0 \sum_{j=1}^{7} \frac{A_j}{s - P_j} \qquad (26)$$

Equation (23) and (24) are a family of linear differential equations with constant coefficients and there is a definite relationship among the values of A_j , P_j , and δk. This relationship can be shown graphically and is presented for reactors of $l^* = 10^{-3}$ sec, 10^{-4} sec, and 10^{-5} sec in Figs.(2-2) and (2-3). Figures (2-2) and (2-3) are used when a positive δk step is involved, and Fig.2-3 only is needed for negative δk steps.

9.1. DEFINITIONS.

1) Block_Diagrams : is a shorthand , graphical representation of a physical system.

To simplify the picture of the complete system,it is common to use a block diagram in which each element in the system is represented by a block.Each block is labeled with the name of the component,and the blocks are appropriately interconnected by the line segments.This type of diagram removes excess detail from the picture and shows the functional operation of the system.

A block diagram represents the flow of information and the functions performed by each component in the system.The simple functional block diagram shows the similarities between different physical systems may be analyzed by the same techniques.

Block diagram consists of a specific configration of four types of elements-blocks,summing points,takeoff-points,and arrows figure.(3-1).

A further step taken to increase the information supplied by the block diagram is to label the input quantity into each block and the output quantity from each block.Arrows are used to show the direction of the flow of the

information.

2) Differential-equation :A property common to all basic laws of physics is that certain fundamental quantities can be defined by numerical values.The physical laws define relationships between these fundamental quantities and are usually represented by equations.

One class of equations which has broad application in the description of physical laws is *differential equations* .

Differential equations are useful for relating rates of change of variables and other parameters.The kinetic response of a system can be understood by examining the differential equation of the system.The differential equation of most linear servo systems is a linear differential equation with constant coefficients of the form of

$$a_n \frac{d^n x}{dt} + a_{n-1} \frac{d^{n-1} x}{dt} \ldots\ldots\ldots a_1 \frac{dx}{dt} + a_0 x = f(t) \qquad (9-1)$$

The solution of this linear constant coefficient ordinary differential equation can be divided into two parts,the steady state response and transient response.The sum of these two responses constitutes the total response,or solution x(t),of the equation.These terms are often used for

specifying control system performance.

3) The Laplace Transform : Once a given order of differential equation has been solved, nothing is obtained in the way of mathematical novelty or physical information by repetitious solutions. Hence it is reasonable that there should be some way of systematizing the process of obtaining the desired information from the equations. The Laplace transform is such tool to do this function.

Symbolically, the Laplace transformation of a function of time to a function of s is written as

$$f(s) = L\,[f(t)] = \int_0^{\infty} f(t)e^{-st}\,dt \qquad (9\text{-}2)$$

and the corresponding inverse Laplace transformation of f(s) back to a function of time is

$$L^{-1}[f(s)] = f(t) \qquad (9\text{-}3)$$

4) Transfer Function :

The ratio of the Laplace transform of the output to the Laplace transform of the input is called the *transfer function*.

$$G(s) = \frac{C(s)}{R(s)} \qquad (9\text{-}4)$$

Commonly, the transfer function is written inside the block, as shown in figure(3-1).

5) Open and Closed loops system :

5.1. Open loop system : system in which the output

quantity has no effect upon the input.

To explain this system we give here an example as shown in figure.(3-2) in which we assume a nuclear reactor having negative temperature coefficient, operating with inlet coolant temperature. So if we take into account control-rod effectiveness we obtain the curve of figure(3-3).

So the output power level is a direct function of the input control-rod position, and we take this curve as a calibration curve of the process.

So in the open-loop system the output will vary as the characteristics of the intermediate components vary as the load changes.

5.2. Closed Loop Systems : System in which the output quantity has an effect on the input.

For example the case of the control of the power output of a nuclear reactor is shown in Fig.3-4. Here the input signal is in the form of a demanded power output from the reactor as measured on a neutron detector. The error signal is amplified and caused to actuate a control-rod drive mechanism. This mechanism in turn positions a control rod in the reactor. If the actual power output is less than the demanded power output, the error signal created is such as to cause the control rod to be slightly withdrawn from the

reactor.When the proper demanded level is reached,the control rod is reinserted to its original position and the error signal disappears.

The advantage of the closed-loop system is that there is little or no dependence upon variations in the primary components of the system.Variations in load have little effect upon system accuracy.In the nuclear-reactor example,the variations in reactor characteristics caused by poison,temperature,or depletion are simply canceled out by control-rod position until the error signal is zero and the output meets the demand .

6) Temperature Coefficients :

Many of the parameters that determine the multiplication factor of a reactor depend on temperature.As a result,a change in the temperature leads to a change in k, and alters the reactivity of the system.This effect has an important bearing on the operation of a reactor and, ultimately, on the safety of the system.

The extent to which the reactivity is affected by changes in temperature is described in terms of the *temperature coefficient of reactivity* ,denoted as α_T.This defined by the relation

$$\alpha_T = \frac{d\rho}{dT} \qquad (9\text{-}5)$$

where ρ is the reactivity and T is the temperature.

Most reactors have negative temperatur coefficient, which means that as the reactor heats up, its reactivity is reduced. Reactors which have water or gas as moderators usually have large negative temp.coefficients. The temperature coefficient will later be shown in this project to be a most important control-system parameter.

2) Depletion :

it is recognized that control rods may have to be moved during the lifetime of a reactor because of fuel depletion. That is as the fuel in the reactor is used up, the number of fissions occuring will decrease; consequently k will be reduced and control rods will have to be moved out to compensate for the reduction in k.

An approximate expression for the reactivity change resulting from fuel depletion is $\delta k \simeq \delta m/2$, where δm is the percentage of the fissionable material burnt up.

3) Fission-Product Poisoning :

All fission products absorb neutrons to some extent, and their accumulation in a reactor tends to reduce its multiplication factor. Since absorption cross sections decrease rapidly with increasing neutron energy, such fission product poisons are of greatest importance in thermal

reactors.

The most important fission product poison is ^{135}Xe, whose thermal absorption cross section is 2.65*10 b and is non 1/v This isotope is formed as the result of the decay of ^{135}I, and also is produced directly in fission. The ^{135}I is also formed in fission and by the decay of ^{135}Te.

These processes and their half-lives are summarized below :

$$^{135}Te \xrightarrow[11\ sec]{\beta^-} {}^{135}I \xrightarrow{\beta^-} {}^{135}Xe \xrightarrow{\beta^-} {}^{135}Cs \xrightarrow{\beta^-} {}^{135}Ba_{(stable)}$$

In view of the fact that ^{135}Te decays so rapidly to ^{135}I, it is possible to assume that all ^{135}I is produced directly in fission. Because the xenon is produced in part by the decay of the iodine concentration. This in turn, is determined by the rate equation

$$\frac{dI}{dt} = \gamma_I \Sigma_f \phi_T - \lambda_I I \qquad (9-6)$$

where ^{135}I = number of atoms of ^{135}I present per cm^3 at any time t

γ_I = is the fractionl yield of ^{135}I .

Σ_f =thermal macroscopic fission cross section of fuel in reactor.

ϕ_T = thermal-neutron flux.

λ_I = decay constant of ^{135}I.

The equation for the concentration of ^{135}Xe in a reactor at any time becomes

$$\frac{dX}{dt} = (\gamma_X \bar{\Sigma}_f - \sigma_X X')\phi_T + \lambda_I I' - \lambda_X X' \qquad (9-7)$$

where X = number of atoms of ^{135}Xe present per cm^3 at any time t.

γ_X = fractional yield of xenon .

σ_X = microscopic thermal-neutron absorption cross section of Xe ($3*10^6$ barns).

λ_X = decay constant of ^{135}Xe .

Two kinetics effects occur which concern reactor control.The first effect is the so-called equilibrium xenon poisoning, and the second the peak xenon poisoning.The equilibrium poisoning occurs during reactor power operation, and the peak poisoning after shutdown of the reactor from a high power level.Figure (3-5) shows the time scale to build up to equilibrium for such a process.The amount of poisoning involved depends upon the steady power level of a reactor and the design of the particular reactor involved.The steady-state xenon concentration may be obtained simply by setting $dX/dt = 0$ in Eq.(9-7) and $dI/dt = 0$ in Eq.(9-6).Then the steady-state concentration becomes

$$\lambda_o = \frac{\Sigma_f(\gamma_X + \gamma_I)\phi_o}{\lambda_X + \sigma_X\phi_o} \qquad (9\text{-}8)$$

From the curve in Fig.(3-6) a very appreciable amount of reactivity may be involved in the peak xenon poisoning.It is conceivable that this amount of reactivity may be so muchh that the reactor does not contain sufficient fuel to completely"override" this peak even when the control rods are pulled out all the way.Under these conditions,where only a fixed amount of reactivity is available,unless the reactor is started up quickly after a shutdown, a large period of time will exist in which it will be impossible to start the reactor until the xenon decays down . As an example from Fig.(3-6),let us assume that sufficient reactivity exists in a reactor that it can still be made critical up to 1/2 hr after shutdown from a given power level. The figure indicates that unless a startup is made within this 1/2 hr,it may be 40 hr before the reactor can be started up again.

9.2. REACTOR TRANSFER FUNCTION :

$$\delta K(s) \rightarrow \boxed{K_R G_R(s)} \rightarrow \delta n(s)$$

Figure(3-7),Block diagram of reactor T.F.

$$K_R G_R(s) = \frac{\delta n(s)}{\delta k(s)} \qquad \text{(Reactor Transfer Function)}$$

1) Reactor Transfer fn.using single-group Delayed n_0^1 :

The reactor kinetics equation for single group of delayed neutrons

$$\frac{dn}{dt} = \frac{\delta k}{l^*} n - \frac{\beta}{l^*} n + \lambda C \qquad (9\text{-}9)$$

$$\frac{dC}{dt} = \frac{\beta}{l^*} n - \lambda C \qquad (9\text{-}10)$$

where C and λ are the average concentration of delayed neutrons and the average value of the decay constant.Now let C as well as n be split up into two parts,a steady state part C_0 and a small axcursion about this steady-state value δC .Then substituting in these new values for n and C ,

$$\frac{dn_0}{dt} + \frac{d\delta n}{dt} = \frac{\delta k n_0}{l^*} + \frac{\delta k \delta n}{l^*} - \frac{dC_0}{dt} - \frac{d\delta C}{dt} \qquad (9\text{-}11)$$

the derivative of a constant is zero and we can ignore the cross products of the two differentials $\delta k \delta n / l^*$ in comparison with $\delta k n_0 / l^*$ .Therefore

$$\frac{d\delta n}{dt} = \frac{\delta k n_0}{l^*} - \frac{d\delta C}{dt} \qquad (9\text{-}12)$$

Similarly ;

$$\frac{dC}{dt} = \frac{dC_0}{dt} + \frac{d\delta C}{dt} = \frac{\beta}{1^*}(n_0 + \delta n) - \lambda(C_0 + \delta C) \qquad (9\text{-}13)$$

and as in the steady state

$$\frac{dC_0}{dt} = 0 = \frac{\beta}{1^*} n_0 - \lambda C_0 \qquad (9\text{-}14)$$

$$\frac{d\delta C}{dt} = \frac{\beta}{1^*} \delta n - \lambda \delta C \qquad (9\text{-}15)$$

Taking the Laplace transforms of Eqs.(9-12) and (9-15)

$$s\delta n(s) = \frac{\delta k(s)}{1^*} n_0 - s\delta C(s) \qquad (9\text{-}16)$$

and

$$s\delta C(s) = \frac{\beta}{1^*} \delta n(s) - \lambda\delta C(s) \qquad (9\text{-}17)$$

The initial-condition transforms have been dropped as we have defined the transform in terms of a steady-state cond.

Combining Eqs.(9-16) and (9-17) gives

$$\frac{\delta n(s)}{\delta k(s)} = \frac{n_0}{s1^*} \frac{1}{1 + \frac{\beta/1^*}{s+\lambda}} = \frac{n_0}{s1^*} \frac{s+\lambda}{s + \lambda + \beta/1^*} \qquad (9\text{-}18)$$

and as λ is generally small compared with $\beta/1^*$ a common approximation is

$$F_R G_R(s) = \frac{\delta n(s)}{\delta k(s)} = \frac{n_0}{s1^*} \frac{(s + \lambda)}{(s + \beta/1^*)} \qquad (9\text{-}19)$$

2) Reactor T.F.-Six Groups of delayed neutrons :

The solution of the reactor kinetic equation for six groups of the delayed neutrons.

By making linearization of n and C into staedy state and small sinusoidal components such that

$n = n_0 + \delta n$; and $C = C_0 + \delta C$ so

$$\frac{dn}{dt} = \frac{\delta k}{l^*} n - \sum_{i=1}^{6} \frac{dC_i}{dt} \qquad (9\text{-}20)$$

because $\sum_{i=1}^{6} \beta_i = \beta$. Then

$$\frac{dn}{dt} = \frac{d\delta n}{dt} = \frac{\delta k}{l^*} n_0 + \frac{\delta k \delta n}{l^*} - \sum_{i=1}^{6} \frac{d\delta C_i}{dt} \qquad (9\text{-}21)$$

$\delta k \delta n / l^*$ may be neglected in comparison with $(\delta k / l^*) n_0$ and

$$\frac{dC_i}{dt} = \frac{d\delta C_i}{dt} = \frac{\beta_i}{l^*} (n_0 + \delta n) - \lambda_i (C_{i0} + \delta C_i) \qquad (9\text{-}22)$$

as in the steady state

$$\frac{d\delta C_{i0}}{dt} = 0 = \frac{\beta_i}{l^*} n_0 - \lambda_i C_{i0} \qquad (9\text{-}23)$$

Equation (9-22) becomes

$$\frac{d\delta C_i}{dt} = \frac{\beta_i}{l^*} \delta n - \lambda_i \delta C_i \qquad (9\text{-}24)$$

Reducing Eqs.(9-21) and (9-24) to Laplace-transform operational form gives

$$s\delta n(s) = \frac{n_0}{l^*} \delta k(s) - s \sum_{i=1}^{6} \delta C_i(s) \qquad (9\text{-}25)$$

$$s\delta C_i(s) = \frac{\beta_i}{l^*} \delta n(s) - \lambda_i \delta C_i(s) \qquad (9\text{-}26)$$

where the initial condition transforms have been dropped.

Combining Eqs.(9-25) and (9-26) gives

$$\frac{\delta n(s)}{\delta k(s)} = \frac{n_0}{l^*} \cdot \frac{1}{s\left[1 + \sum_{i=1}^{6} \frac{\beta_i}{l^* (s+\lambda_i)} \right]} \qquad (9-27)$$

which in expanded form becomes

$$\frac{\delta n(s)}{\delta k(s)} = \frac{n_0}{l^* s} * \frac{(s+\lambda_1)(s+\lambda_2)......(s+\lambda_i)}{(s+\lambda_1)(s+\lambda_2)...(s+\lambda_i) + \frac{\beta_i}{l^*}(s+\lambda_2)(s+\lambda_3)(s+\lambda_i) + \frac{\beta_2}{l^*}(s+\lambda_1)(s+\lambda_3)....(s+\lambda_i) + ..} \qquad (9-28)$$

In order ultimately to obtain the form ..

$$\frac{\delta n(s)}{\delta k(s)} = \frac{n_0}{l^* s} \cdot \frac{\prod_{i=1}^{6} (s+\lambda_i)}{\prod_{i=1} (s+r_i)} \qquad (9-29)$$

3) Approximation Reactor Transfer function :

The six-group delayed-neutron formulation of the reactor transfer function although exact, is not easy to manipulate in later analytical work. Approximate formulas using some form of average values of constants are satisfactory in many problems. the method of obtaining average constants in some cases depends on the problem. Equation(9-19) gives an easy form of approximation to use, with the question being, What is a good value for an average λ and β/l^* ? So we assume an approximate transfer function of the form

$$k_R G_R(s) = \frac{n_o (s+\bar{\lambda})}{l^* s (s+\bar{r})} \qquad (9\text{-}30)$$

the appropriate engineering constants might be $\bar{\lambda} = 0.125$ sec and $\bar{r} = 50$ sec even though these numbers have no direct physical meaning.This approximation is plotted in comparison with the exact expression in Fig.(3-7).

9.3. REACTOR T.F. WITH TEMP. COEFFICIENT FEEDBACK :

As shown in Fig.(3-8) the combination transfer function will be :

$$k_{RTc} G_{RTc}(s) = \frac{k_R G_R(s)}{1 + k_R G_R(s) k_{Tc} G_{Tc}(s)} \qquad (9\text{-}31)$$

where $k_{RTc} G_{RTc}(s)$ is the new combined over-all transfer function and $k_{Tc} G_{Tc}(s)$ is the transfer function of the local temperature-coefficient effect.

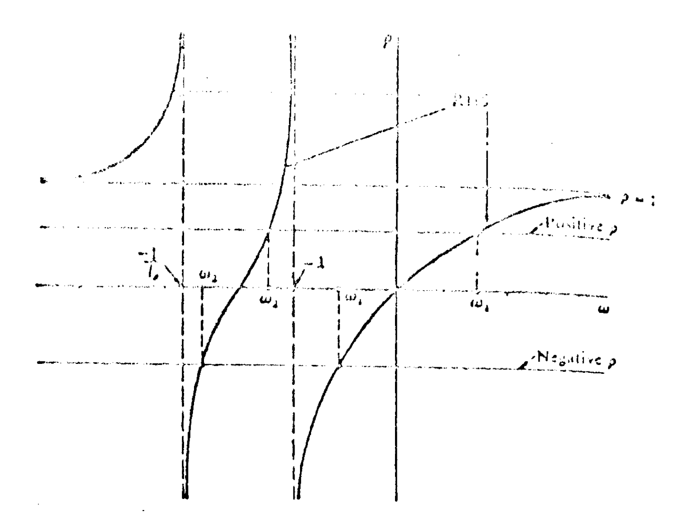


Fig. 1 Plot of the reactivity equation for one group of delayed neutrons. (Not to scale.)

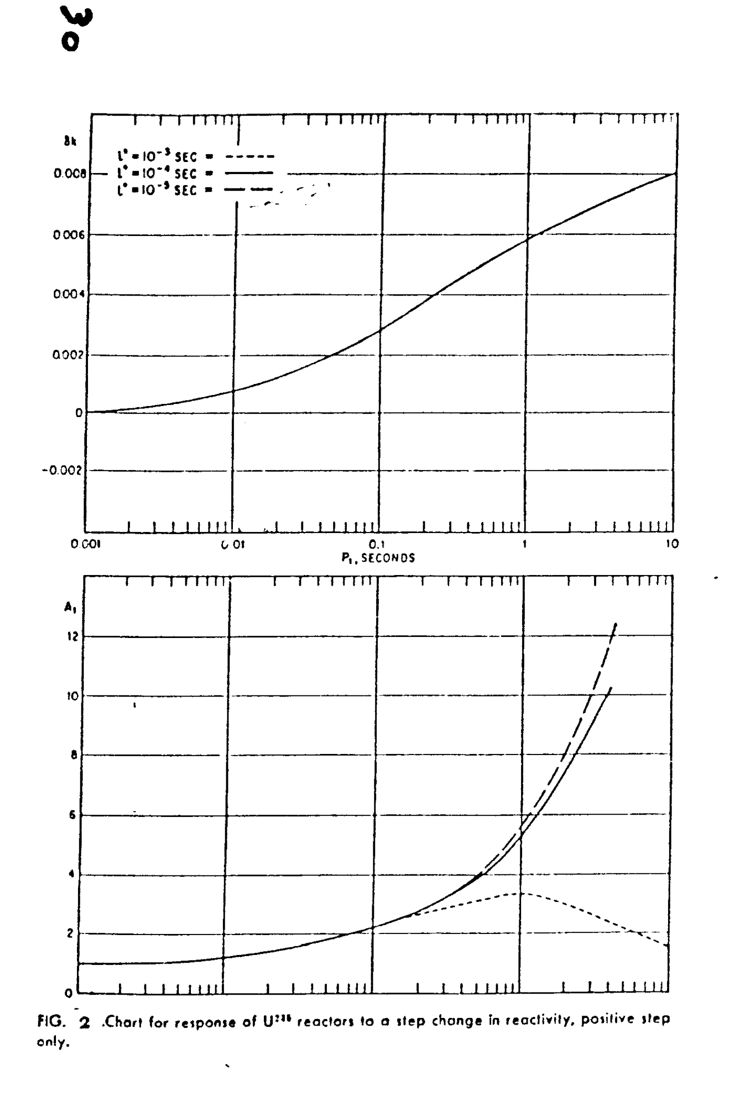


FIG. 2 Chart for response of U^{235} reactors to a step change in reactivity, positive step only.

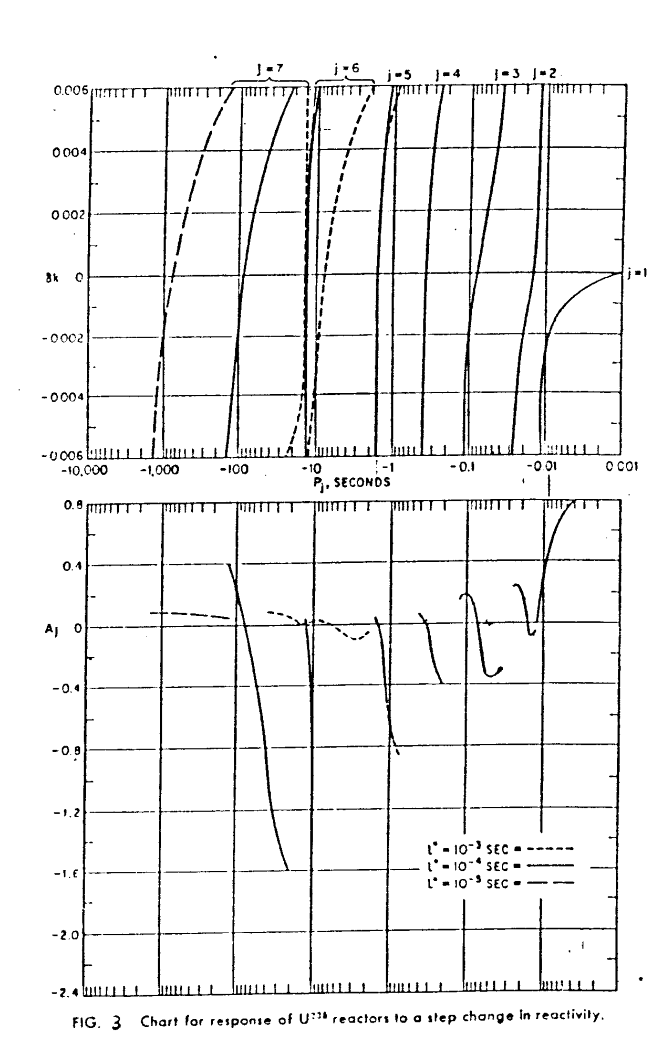
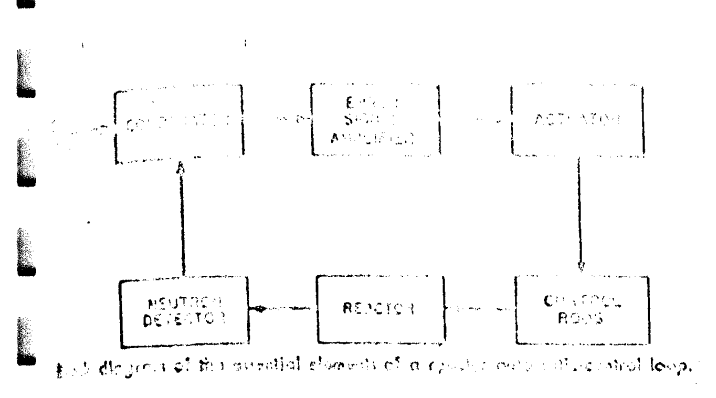


FIG. 3 Chart for response of U^{235} reactors to a step change in reactivity.



[illegible] diagram of the essential elements of a [illegible] control loop.

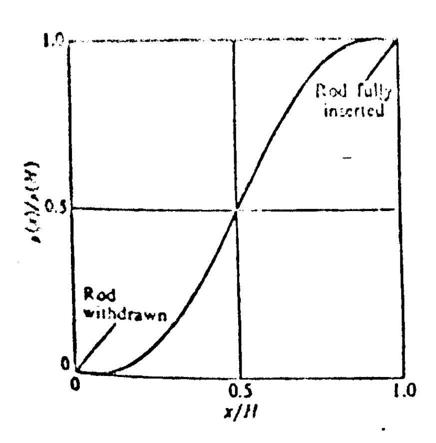


Fig. 5 The worth of a partially inserted control rod as a function of the distance of insertion.

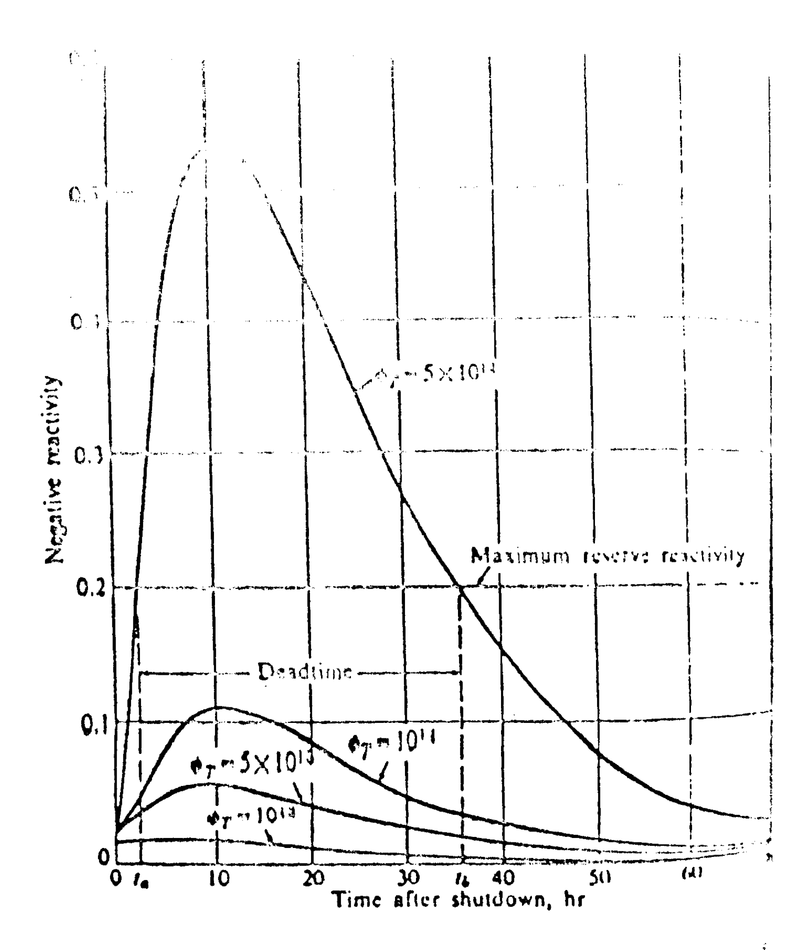


Fig. 6 Xenon-135 buildup after shutdown for several values of the operating flux before shutdown.

17

3

31

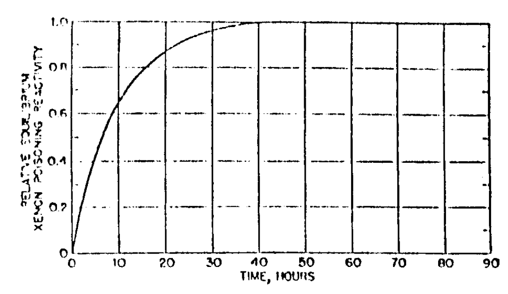


FIG. 7 . Equilibrium xenon poisoning build-up for an enriched thermal reactor.

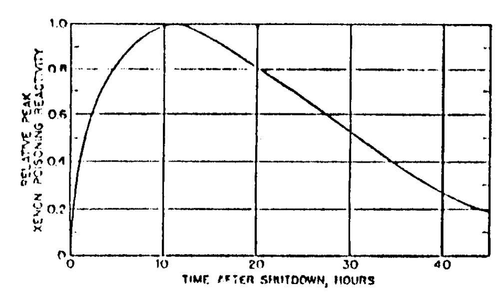


FIG. 8 Relative peak xenon poisoning reactivity as a function of time after shutdown for a thermal reactor.

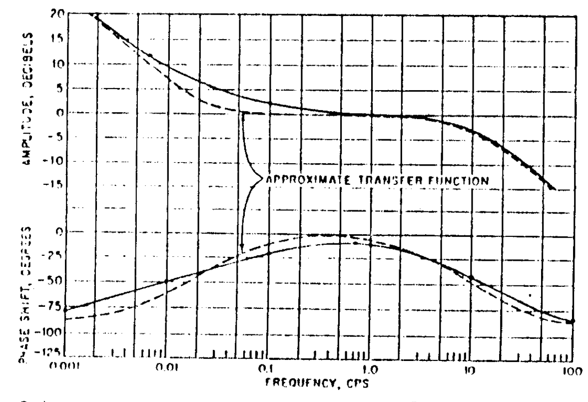


FIG. 9 . Exact transfer function and approximation when $\bar{\lambda} = 0.125$ and $\bar{\tau} = 50$ for a U^{235}-fueled reactor.

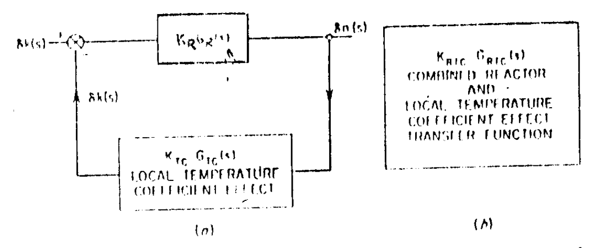


FIG. 10 Combination of reactor transfer function with local temperature coefficient feedback transfer function. (a) Individual transfer functions. (b) Combination transfer function.

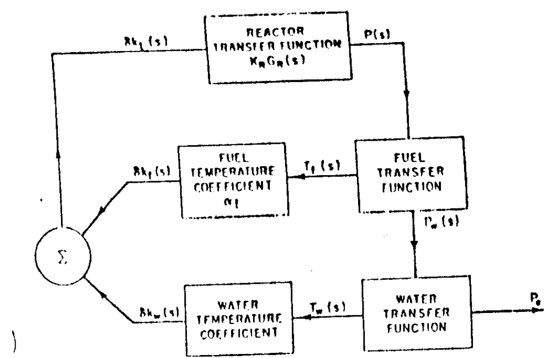


FIG. 11 . Block diagram of two-loop feedback system.

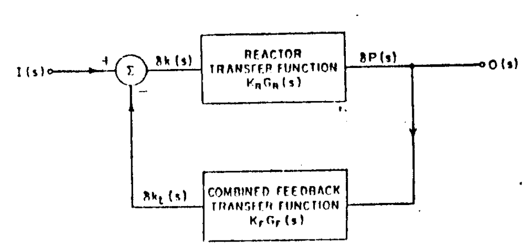


FIG. 12 . Simplified block diagram of reactor and over-all temperature feedback.

Nuclear Reactor tinetics

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Lec. 4.1

Reactor Heat Transfer and Fluid Flow

Mohamed Khattab

- *Core Thermal-hydraulics*
- *Core Pressure Drop*
- *Transient Analysis*

HIGHEST ADMISSIBLE OPERATING POWER OF ET-RR-1* CORE

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ABSTRACT

ET-RR-1 is 2 Mw nuclear research reactor. In order to have higher neutron flux for experimental purposes and isotopes production rising power is investigated.

Difference between in-pile measurements and the highest allowable clad temperature is about 35 C at nominal operating conditions.

Core configuration, bundle dimensions and reactor operating conditions are introduced as input data into computer program to predict core thermohydraulic pattern and clad temperatures.

Evaluation model shows that at 860 m^3 /hr. the maximum allowable operating power is 2 Mw.

It is possible to rise power up to 4 Mw by increasing coolant flow rate to 2100 m^3 /hr. without affecting clad integrity.

KEYWORDS

Reactor power, ET-RR-1 core, power increase, thermohydraulic pattern, flow distribution.

INTRODUCTION

In order to increase neutron flux for experimental purposes and isotopes production, it is necessary to rise reactor power.

ET-RR-1 is water cooled reactor of 2 Mw, 860 m^3/hr. nominal operating condition (1). Briefly, the core contains 51 locations, Fig. 1, for loading fuel bundles or dispalcers. Each bundle contains 16 fuel rods , type EK-10, aluminum clad, 1 cm outer diameter. Displacers are used for isotopes production and flow management.

Due to the presence of control rods four bundles have three cut-corners, eight have two cut-corners, eight have one cut-corner and the rest are squares. Bundle side length is 6.3 cm, recess of cut-corner is 1.8 cm.

In-core measured clad temperatures did not exceed 55 C at 2 Mw and 860 m^3/hr.(2), (3). Highest allowable clad temperature specified by the vendor is 90 C to maintain clad integrity (4). The 90 C is less than

* Egypt Research Reactor -1

coolant boiling point and to avoid errosion and corrosion of the Aluminum clad. The difference between measurements and highest allowable clad temperature, 35 C, encourages the idea of investigating reactor power increase without changing core configuration.

The highest admissible operating power of ET-RR-1 core from thermohydraulic point of view is investigated in this work.

CORE THERMOHYDRAULIC PATTERN

Reactor power level and thermal load of each fuel element is one of the most important factors upon clad temperature.

In the present work, the core is divided into seven groups to adapt radial power distribution. Power generated per bundle per group at 2 Mw for Bessel function first kind, zero order, $J_o(2.4048\ r/R)$, is shown in Fig. 2. The core contains 41 fuel bundles. The seventh group consists of 10 displacers, without fuel rods. Relative position and power per bundle are given in table 1.

In supporting plate underneath the bundles, orifices of diameters 4.5, 3.25 or 2.5 cm are fitted to distribute the coolant to get best thermal performance. Measured pressure drop coefficients are used to calculate flow distribution. Effect of cut corners on flow rate per bundle is insignificant (5). Flow rate per bundle is given in table 1. Flow rate per bundle complies with orifice diameter.

Table 1 Power and coolant per bundle at 2 Mw, 860 m^3 /hr.

Group	Number of bundles	Relative position	Power/bundle Kw.	Flow/bundle m^3 /hr.	Orifice cm.
1	4	0.125	92.39	36.41	4.5
2	8	0.375	76.25	21.38	3.25
3	4	0.530	59.85	21.38	3.25
4	8	0.625	48.22	13.08	2.5
5	8	0.750	31.94	13.0	2.5
6	9	0.875	15.53	13.08	2.5
7	10	1.000	0.00	13.08	2.5

Due to fuel depletion with time some displacers may be replaced by fuel bundles. The flow distribution will not be affected, also the variation in power distribution is insignificant.

CLAD TEMPERATURES

Clad temperatures are calculated through flexible program surveying the core. The following concepts are taken into consideration :

- sinus axial power generation with extrapolated length effect;
- effects of control rods and fuel depletions on power distribution are neglected;

- power per bundle is uniformly distributed between its rods;
- single phase turbulent flow heat transfer correlation is;

$$Nu = C\, Re^{.8}\, Pr^{.4}$$

- equivalent diameter depends on bundle shape;
- at the mean bundle coolant axial temperature, physical properties are calculated for each group;
- coolant inlet temperature is 34 C.

The constant of heat transfer correlation , C=0.023 is used in nuclear reactors calculations as design criteria for Evaluation Model (EM) to achieve safety conditions.

Bundle average coolant velocity and its equivalent diameter are used to calculate Reynolds number. Square bundle D_e is 1.44 cm. Bundles of one, two and three cut corners have D_e 1.37, 1.30 and 1.23 cm. respectively.

Maximum calculated clad temperatures using C=0.023 for 2 Mw, 860 m^3 /hr. are given in table 2 and shown in Fig. 3. Core highest clad temperature approaches 78 C in bundles of second and fourth group due to interactions of coolant and power distributions.

Flat radial power distribution corresponds to 48.8 Kw./bundle, leads to highest clad temperature of 81 C in bundles of 4,5 and 6 group. Clad maximum temperature position does not exceed 1 cm far from midpoint of fuel rod.

Table 2 Maximum clad temperatures at 2 Mw, 860 m^3 /hr.

Group	Re	Heat transfer coeff. w/cm² C	Maximum clad temperature C	Coolant rise C
1	77600	1.786	66.4	2.2
2	45300	1.095	77.7	3.1
3	43700	0.967	72.7	2.4
4	27400	0.695	77.6	3.2
5	26700	0.652	64.7	2.1
6	26400	0.648	49.0	1.0

Variations in Reynolds number is less than 2% for bundles of groups 4, 5 and 6 , where flow rate per bundle is 13.08 m^3 /hr., due to cut corners. The same effect takes place on the second and third groups, 21.38 m^3 /hr.

Two scram cases for reactor safety operations (4) are :
- 20 % power increase.
- 10 % reduction in flow rate.

At power increase limit, highest clad temperature is 87 C. Coolant reduction limit leads to 82 C. Applying the two conditions simultaneously highest clad temperature reaches 91 C. This value reaches cladding temperature integrity limit.

Hence, it is evident that 2 Mw represents the maximum allowable operating power at 860 m^3/hr.

Better fuel rods coolability could be achieved by preventing flow through displacers. The flow rate through bundles increase and clad temperatures decrease by about 5 C only.

POWER INCREASE

Number of fuel bundles in reactor core is constrained by self sustained chain reaction control, irrespective of operating power. Power is limited by coolant capability to transfer the generated heat from fuel elements.

In-pile clad temperature measurements analysis show that a value of C = 0.047 agrees satisfactorily with the measurements (6). This value could be considered as that of Best estimate Model (BM). Calculated maximum clad temperature in core is 56.2 C, Fig. 3.

The temperatures at first, second and both emergency margins will be 61, 59 and 63 C respectively.

Highest clad temperature of flat radial power distribution is 58 C.

The relation between flow rate and operating power keeping axial clad-coolant temperature difference, neglecting physical properties temperature dependence, is

$$F = (P/P_n)^{1.25} \quad F_n$$

Core average coolant temperature rise in this case is

$$T = (P/P_n) \quad T_n^{-.25}$$

Fig. 4 shows the variation of highest clad temperature with total core coolant flow rate at doubling power. It is clear from EM that fuel bundles can generate 4 Mw without exceeding the allowable clad temperature limit at 1500 m^3/hr. To permit for simultaneous safety margins at 4 Mw coolant flow rate must be increased to 2100 m^3/hr.

From BM the highest clad temperature is 79 C, at 4 Mw and 860 m^3/hr., while it is 91 C at simultaneous safety margins, keeping fuel bundle structure unchanged. This model although it is promising it must be verified through extensive experimental work and approved by safety committee.

In Poland, nominal reactor power of EWA-2 was doubled to 4 Mw without exceeding clad temperature limit, keeping core flow rate at 910 m^3/hr. (7). The EK-10 fuel rods were placed in venturi tubes in order to increase water velocity around fuel rods. Rising power up to 8 Mw was achieved by using fuel clusters of three concentric fuel tubes, outer one being hexagonal, at coolant flow rate 1200 m^3 /hr.(8).

CONCLUSIONS

To keep cladding material integrity, evaluation model shows that at 860 m^3 /hr. coolant flow rate, the maximum allowable operating power is 2 Mw.

It is necessary to have about twice and half nominal flow rate to keep the highest clad temperature within permissible limits at 4 Mw without changing design of bundles. Structure stability of fuel bundles must be tested at new flow condition.

According to best estimate model, BM, calculated highest clad temperature at 4 Mw and 860 m^3/hr. is 79 C. Since BM is not approved by safety authorities in nuclear reactors design, doubling nominal reactor power without increasing coolant flow rate or changing design of fuel bundles is not advisable.

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Nomenclature

D_e equivalent diameter;
Nu Nusselt number;
Re Reynolds number;
Pr Prandtl number;
F core coolant flow rate;
P reactor power;
T temperature difference.
subscripts
n nominal

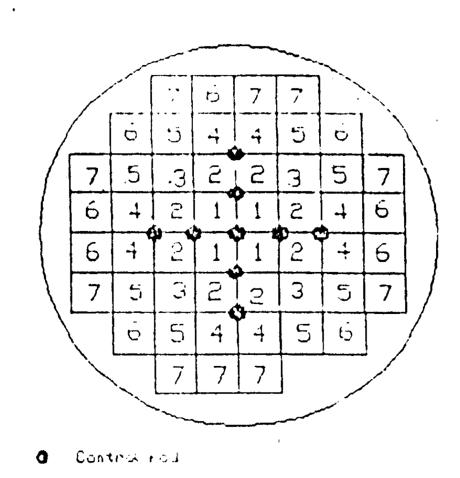


Figure (1) Core regions

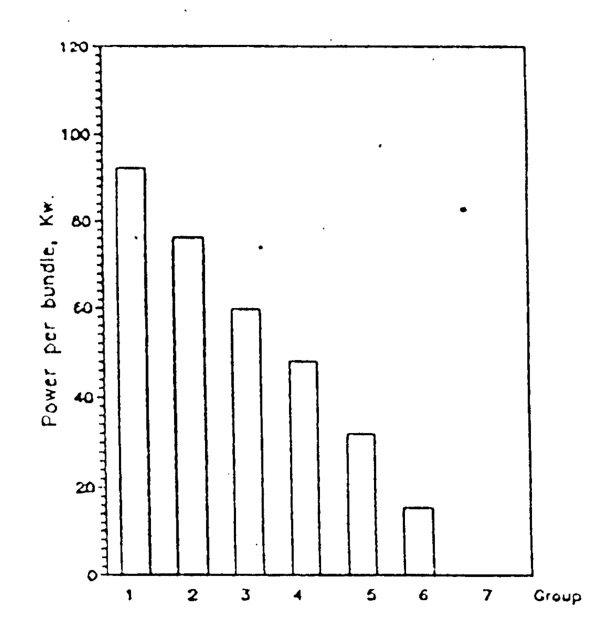


Figure (2) Power per bundle at 2MW.

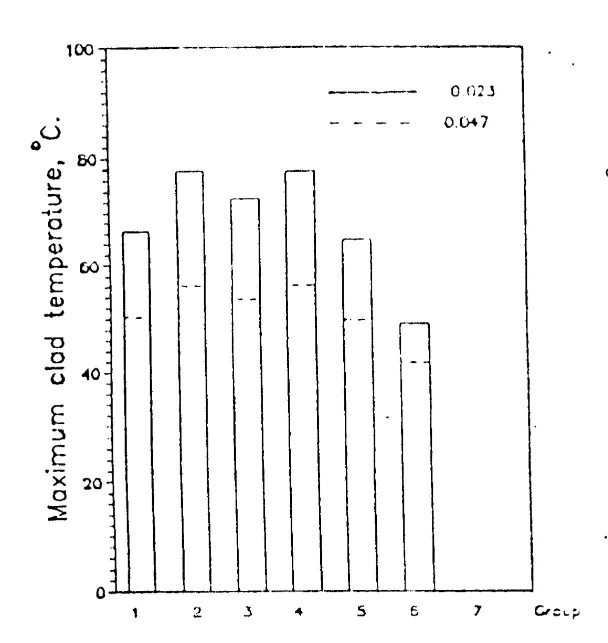


Figure (3) Maximum clad temperature, °C at 800 M^3/hr. and 2MW.

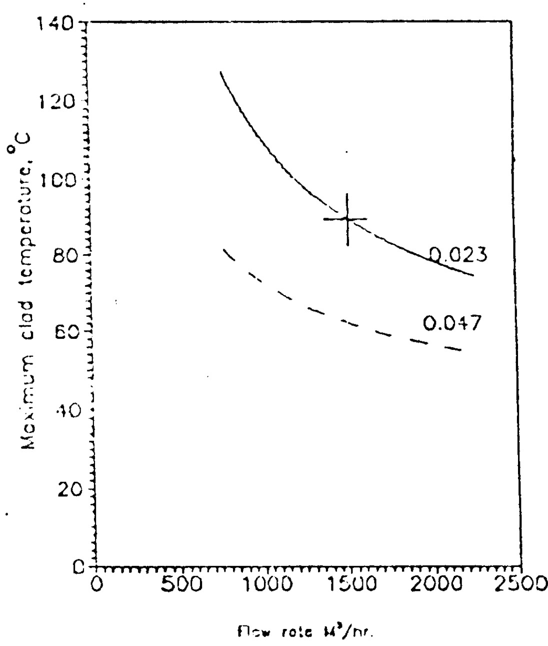


Figure (4) Effect of flow rate on maximum clad temperature at 4 MW.

Fourth International Conference of Fluid Mechanics

ICFM4 - Alexandria April 28 - 30, 1992

INVESTIGATION OF PRESSURE DROP THROUGH FUEL BUNDLES IN RESEARCH REACTORS

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Abstract

In nuclear reactors coolant may be arranged through individual bundles by fitting particular washers to achieve better thermal performance. Pressure drop through a bundle comprising 16 rods, grids and washers similar to those of Egypt research reactor (ET-RR-1) core is measured and flow distribution is predicted.

Experiments are carried out under adiabatic conditions using water at about 35 C. Reynolds number range is $4x10^4$ $-2x10^5$. Washers of one centimeter thickness and inside diameters 4.5, 3.25 and 2.5 cm are tested.

The results show that bundle and lower grid pressure drop coefficients are 3.75 and 1.8. Washers pressure drop coefficients are 2.65, 19.67 and 53.55 respectively.

A model based on constant pressure drop across bundles in the core is developed to determine the flow distribution. Application of measured pressure drop coefficients leads to 39.1, 20.4 and 13.1 m^3/hr per fuel bundle depending on washer diameter. The total pressure drop is about 62 kPa.

1- INTRODUCTION

Flow rate per bundle is needed to perform core thermal-hydraulic analysis. Coolant distribution depends on pressure drop through individual bundles.

To achieve better thermal performance coolant may be distributed through fuel bundles by fitting washers of different internal diameters under the bundles [1].

In the present experimental activity washers and fuel assemblies pressure drop are measured under hydraulic and geometric conditions closely similar to those prevailing in ET-RR-1 [2]. The core is provided with washers of internal diameters 4.5,3.25 and 2.5 cm, as shown in figure (2). The total flow rate at normal operating conditions is 860 m^3/hr.

Previous work investigated pressure drop due to friction in rod bundles. Grillo and Marinelli [3] studied friction pressure drop in single and two phase flow on a 16 rod, 1.5 cm diameter, lattice pitch 1.9 cm in square bundle. They showed that for single phase flow the friction factor is less than that predicted by Moody's correlation for smooth tubes by about 10%.

Tong [4] stated that Berringer found for 36 rods bundle, 0.9 cm diameter, pitch to diameter ratio 1.27, friction pressure drop was about 15% higher than that calculated by Moody's correlation. Reynolds number varies from 10,000 to 30,000.

The objective of the present work is to determine pressure drop coefficients of bundle, grids and washers. These coefficients are used to determine coolant flow rate distribution at same total pressure drop across the core. Total pressure drop consists of pressure drop across the washer and that through the bundle. The last one is due to friction along the bundle and pressure drop across upper and lower grids which are used for positioning and fixation of rods in the bundle.

2- EXPERIMENTAL APPARATUS

Figure (1) is a schematic diagram of the experimental loop. The tested bundle is square, 6.8 cm side, with simulated fuel rods assembled between two grids. The tested bundle has 16 aluminum rods, (1 cm diameter ,60 cm length), arranged in 1.7 cm square pitch. The assembling lower and upper grids thickness are 1.4 cm and 0.8 cm respectively. The upper grid has a special bridge for bundle manipulation. The flow area between the rods is 27.12 cm^2. The lower end of the bundle is circular of 5.6 cm diameter. To avoid entrance effect, a square bundle without rods nor lower circular part is mounted above the tested bundle.

Pressure drop through the supporting lower grid is determined from measurements across a simulated riveted rod position grid mounted at the middle of the guide bundle.

The flow rate is measured by a calibrated orifice flow meter. The maximum measured water flow rate is 20 m^3/hr. The pressure drop across the grid, tested bundle and washers are measured by mercury tubes.. The measurements represent directly the pressure drop since the static head is compensated. Measuring taps diameter is 0.2 cm. Small values of pressure drop are measured by using inclined tubes.

3- PRESSURE DROP ACROSS THE CORE

The pressure drop across the core is composed of two components :

3.1- Pressure drop across the washer

Considering one dimensional flow ,momentum conservation and form drag pressure drop [4]

$$\Delta P_w = K_w \rho V_e^2 /2 \qquad (1)$$

The coefficient K_w depends on washer inside diameter.

3.2- Pressure drop across the bundle

Pressure drop across the bundle, ΔP_b, consists of that of the lower and upper grids, ΔP_g, and friction along the bundle, ΔP_f

$$\Delta P_g = K_g \rho V_e^2 /2 \qquad (2)$$

Single phase friction pressure drop along the bundle is generally calculated by Darcy formula.

$$\Delta P_f = f L \rho V^2 /2D \qquad (3)$$

The friction factor, f, for turbulent flow is proportional to Reynolds number to exponent -0.2. The variation of f in our experimental range is small, therefore, equation (3) takes the form

$$\Delta P_f = K_f \rho V_e^2 /2$$

$$\Delta P_b = \Delta P_g + \Delta P_f$$

$$= (K_g + K_f) \rho V_e^2 /2$$

$$= r \rho V_e^2 /2$$

$$\Delta P_t = \Delta P_b + \Delta P_w$$

$$= (r + K_w) \rho V_e^2 /2$$

$$= K_j \rho V_e^2 /2 \qquad (4)$$

where j equals 1,2 and 3 corresponding to washer inside diameter.

4- RESULTS

The velocity at bundle exit, 5.6 cm. diameter, is taken as reference to calculate the pressure drop coefficients. The average velocity between rods in the square bundle is 0.73 that of bundle exit.

Pressure drop through bundle, 4.5 cm washer and simulated grid are presented in figure (3). Bundle pressure drop is greater than that through the washer. The ratio of bundle pressure drop to that of 4.5 cm washer diameter is 1.415. Bundle pressure drop coefficient, r, is 3.75.

The simulated grid pressure drop coefficient is 1.8 which is 48 % of bundle. The calculated friction pressure drop using Moody's correlation at the experimental conditions is less than 50 Pa. which is less than 1 % of the measured bundle pressure drop.

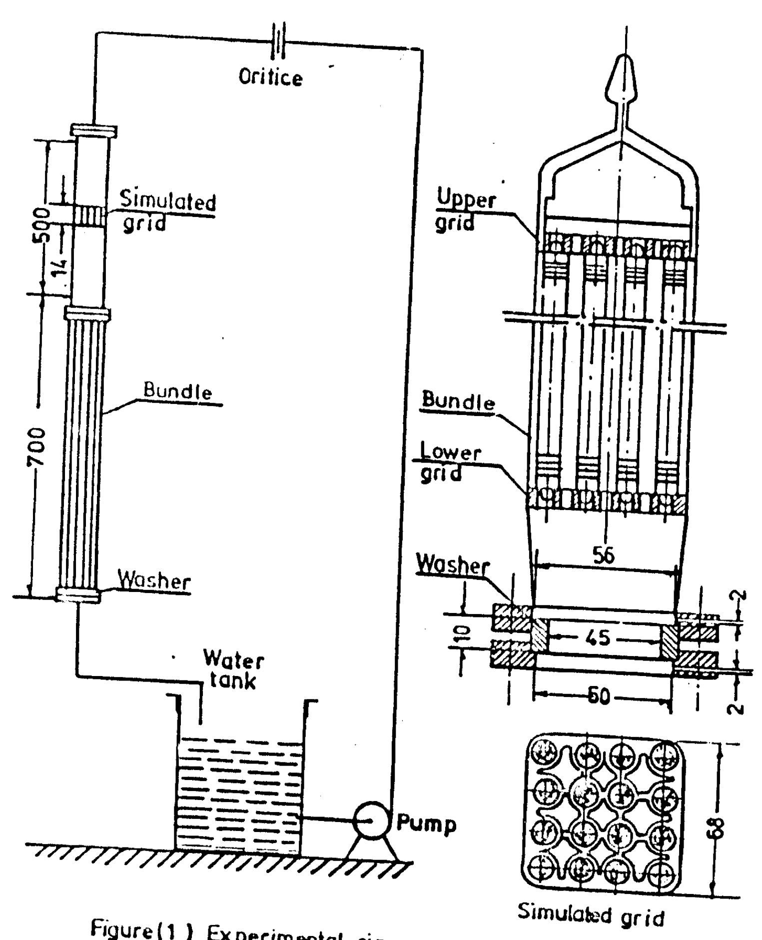
Nomenclature

f friction factor;
K pressure drop coefficient;
L length; m.
M total flow rate; m^3/hr
N number of bundles;
V velocity; m/sec.
ΔP pressure drop; Pa.
r bundle pressure drop coefficient;
ρ density, kg/m^3
$\emptyset$ ratio of velocities

Subcripts
b bundle,
e exit,
f friction,
g grids,
t total,
w washer,

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Figure(1) Experimental rig. (Dim in.mm.)

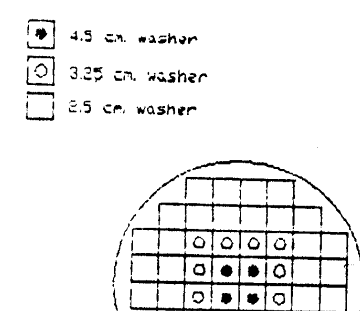


Figure (2) Washers distribution in ET-RR-1

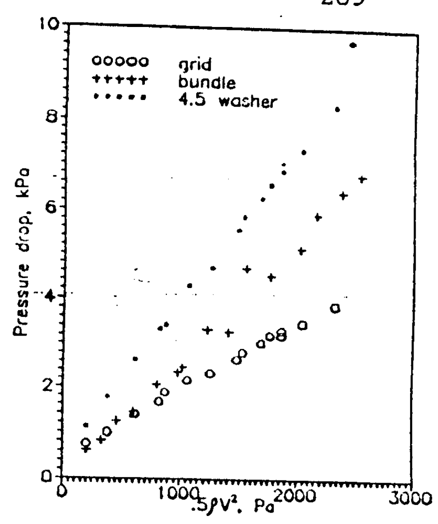


Figure (3) Grid, bundle and 4.5 washer pressure drops.

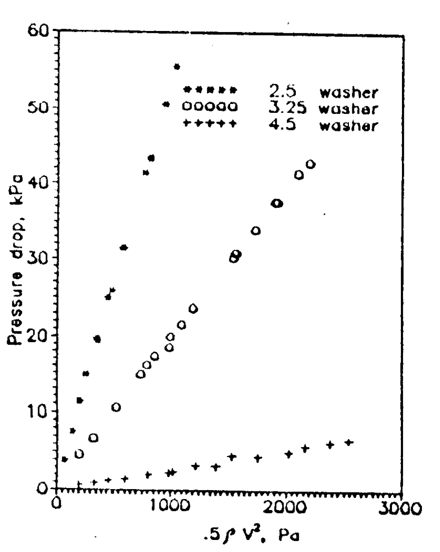


Figure (4) Presuure drop through washers.

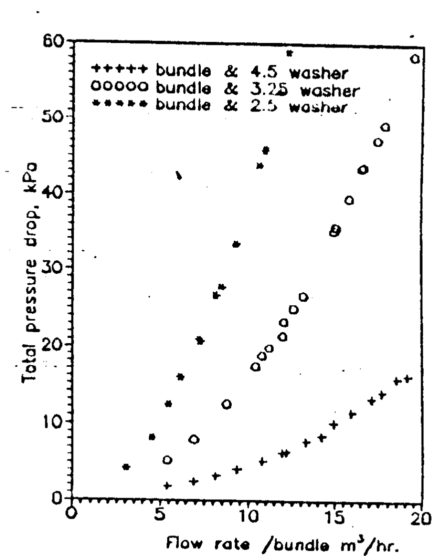


Figure (5) Total pressure drop.

TRANSIENT FUEL TEMPERATURE IN RESEARCH REACTORS

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Abstract

Numerical method is described to solve heat diffusion equation in both fuel and clad materials during and after shutdown. Clad surface temperature is calculated and verified with measured in-pile data at scram and normal shutdown corresponding to periods of 0,05 and 10 sec. respectively. Fuel maximum attainable temperature during transients is determined. Fuel temperature does not increase considerably whereas reactor tripping takes place within 10 % flow reduction irrespective of flow coast down time.

1. INTRODUCTION

Reactor power decreases exponentially during shutdown due to negative reactivity depending on control rod insertion rate. After shutdown, the power continues to decay due to residual fission and radioactive decay of fission products. Decay mode is dominated by half life of the longest lived radioactive product.

Fuel rod cladding temperatures in transient states were measured for maximum credibility in reactor operation incidents, i.e. cooling water flow failure and reactor power rise as a result of ramp reactivity addition [1]. Flow coast down from 100 % to 0 % takes place within 2 seconds in 10 MW research reactor [2], by measuring pressure after pumps.

Measurements of cladding surface temperatures were performed during reactor scram and normal power shutdown as well after pumps switching off [3].

Theoretical temperature distribution are obtained for fuel and coolant when axial heat generation in fuel increases exponentially with time without taking into consideration the transversal heat conduction in fuel [4]. The analytical solution involving convolution integrals and special functions containing infinite series while current practice for such problems is tackled numerically.

In present work, heat diffusion equation in one dimensional cylindrical coordinates has been solved numerically taking into consideration the heat source during and after shutdown in reactor fuel. The solution is verified with experimental data. Incidental coolant decrease is also investigated.

2. GOVERNING EQUATIONS

Heat diffusion equation with internal heat generation in solids having constant thermal properties in axisymetric cylindrical coordinates neglecting axial temperature gradients is [5], [6]

$$\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r} + \frac{\dot{q}}{K} = \frac{1}{\alpha}\frac{\partial T}{\partial t} \qquad (1)$$

where $T \ni T(r,t)$

Clad material heat diffusion equation is similar to eqn. (1) without the heat source term.i.e.

$$\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r} = \frac{1}{\alpha_c}\frac{\partial T}{\partial t} \qquad (2)$$

Explicit finite difference method is used to solve the above two equations in both fuel and cladding materials at the following boundary conditions.

$$\frac{\partial T}{\partial r} = 0 \qquad \text{at } r = 0 \qquad (3)$$

$$K_f \frac{\partial T}{\partial r} = K_c \frac{\partial T}{\partial r} \qquad \text{at } r = \text{fuel clad interface} \qquad (4)$$

$$-K_c \frac{\partial T}{\partial r} = h(T_c - T_s) \qquad \text{at clad coolant interface} \qquad (5)$$

Time derivative is approximated by forward difference scheme with first-order truncation error as follows :

$$\frac{\partial T}{\partial t} = \frac{1}{\Delta t}\left[T_i^{n+1} - T_i^{n}\right] \qquad (6)$$

Space derivatives are approximated by

$$\frac{\partial T}{\partial r} = \frac{1}{\Delta r}\left[T_{i+1}^{n} - T_i^{n}\right] \qquad (7)$$

$$\frac{\partial^2 T}{\partial r^2} = \frac{1}{\Delta r^2}\left[T_{i+1}^{n} + T_{i-1}^{n} - 2\,T_i^{n}\right] \qquad (8)$$

The space derivatives given in eqns. (7) and (8) are evaluated at time step n. Radial position, r, is taken at nodes center .

To solve eqns. (1) and (2) numerically fuel radius and clad thickness are divided into NRF, number of regions in fuel, and NRC, number of regions in clad , respectively, Fig. 1. This leads to

$$\Delta r_f = \text{fuel radius} / NRF$$

$$\Delta r_c = \text{clad thickness} / NRC$$

To achieve stability condition, for one dimensional explicit finite difference numerical solution, time step is constrained by [7]

$$\alpha \Delta t / \Delta r^2 < 0.5 \qquad (9)$$

3. COMPUTER PROGRAM AND CALCULATIONS

Computer program is written to solve simultaneously eqns. (1) and (2). Input data are, fuel radius, NRF, K_f, α_f, clad thickness, NRC, α_c, initial values of $\dot{q}$, T_s, h for forced and natural convection, shutdown per, power percentage at end of exponential decay interval, flow coast down time, scram delay, coolant temperature after shutdown, t, number of time steps and number of time steps after pumps switching off.

Heat Source Term

The following three modes of volumetric heat source, $\dot{q}$, in eqn.(1) are considered, neglecting reactor operating time before shutdown:

a- During shutdown,i.e. during control rods insertion, the power is decaying exponentially according to the relation

$$\dot{q} / \dot{q}_o = \exp(-t/per) \qquad (10)$$

where per is the period of power decrease,i.e. the time required to decrease power to 1/e or 36.8% of its initial value. Immediately after shutdown about 7 % of energy release is produced. Within one or two seconds this value reaches 3 % due to fission products decay [8].

b- Interval between shutdown and pumps switching off, the power is kept constant equal to that at the end of exponential decay.

c- After pumps switching off, the power is again considered varying with time by the following relation [9].

$$\dot{q} / \dot{q}_o = A \; t^{(-0.26)} \qquad (11)$$

where A = power ratio at end of exponential decay. In eqn. (11) the time ,t, starts with 1 sec. i.e without time accumulation of previous intervals.

Different volumetric heat source terms described above are shown in Fig. 2.

Heat Transfer Coefficient

Forced convection heat transfer coefficient at normal operating conditions is introduced into the program as input data. Initial natural convection heat transfer coefficient after pumps switching off is also introduced as input data and variation with decaying power is calculated by the program. In case of flow coast down the program calculates heat transfer coefficient corresponding to instantaneous flow rate and bounded by natural convection value.

Coolant temperature

Coolant temperature varies linearly after 5 sec. delay from initial T_S to coolant temperature after shutdown within 4*(per+4) sec. Initial temperature distributions in clad and fuel are calculated from steady state solution of eqns. (2) and (1). Transient calculation starts by checking stability conditions eqn.(9). In case of instability, Δr or Δt is changed.

4. VERIFICATION AND APPLICATIONS

The solution is verified with measurements at scram and normal shutdown modes of EK-10 fuel in Egypt Research Reactor [3]. Forced convection heat transfer coefficient and initial generating power in the bundle are 10 $kW/m^2 \,°C$ and 41.5 kW. respectively [10],[11].Coolant initial temperature is 32 °C. Exponential power decay ends at 0.03.

From eqn.(10) after 3.5 periods power decreases to about 3% ,hence the period is taken to be 1/3.5 the time of control rods insertion in calculations.

- Scram Mode

In scram mode, safety rods insertion time is 0.18 sec. Based on a period of 0.05 sec.in eqn. (10), clad surface temperature drops from 55 °C to about 27 °C in about 20 sec., Fig.(3).

Normal Shutdown Mode

In normal shutdown, time of control rods descending from operating position is about 35 sec. Based on a period of 10 sec., clad surface temperature drops from 55°C to about 27 in about 60 sec.

Fig. 3. shows good agreement between expected and in-pile measurements [3] in both cases of scram and normal shutdown.

Temperature after pumps switching off

Transient calculation after pumps switching off starts with the calculated temperature distribution in fuel after shutdown as initial values. The power is decaying according to equation (11). Heat transfer is governed by natural convection. In this condition heat transfer coefficient is 0.3 $kW/m^2 °C$ [12].

Fig. 4. shows good agreement between model results and experimental data. The clad temperature rise is due to the variation in heat transfer mode from forced to natural convection. The temperature rise is about 5 °C. As power decays insignificantly with time, eqn. (11), natural heat transfer coefficient remains unchanging also, consequently the small temperature difference will persists .

The following two cases are investigated :

- In hottest fuel bundle, heat transfer coefficient and generated power are 17 kW/m^2 C and 65.6 kW respectively [9].

Transient temperatures during and after shutdown of fuel , clad and coolant are shown in Fig. 5. Initial maximum fuel center, clad surface and coolant temperatures are about 100°C, 56 C and 32°C respectively. They fall to 28 °C , 26 °C and 25°C within 60 sec. In the figure cooling interval between shutdown and pumps switching off is compressed where the temperatures are nearly constants. Fuel and clad temperatures rise about 10°C after pumps switching off.

- Incidental coolant decrease during normal operation causes automatic scram at certain pre-set flow reduction to maintain fuel integrity and reactor safety. Flow coast down takes place in certain interval depending on the cause and system. In case of flow coast down within 2 sec.,two pre-set cases of 10 % and 50 % flow reduction are studied and shown in Fig. 6. These pre-set values correspond to starting scram after 0.2 and 1 sec. respectively from the incident initiation. Fuel center and clad reach about 115°C at 0.2 sec scram delay and 165 C at 1 sec.

Case of flow coast down within 10 sec. at scram delays of 1 and 5 sec. corresponding to 10 % and 50 % flow reduction respectively are shown in Fig. 7. In this case due to long delay before scram temperatures reach about 300 and 100 °C .

5. CONCLUSIONS

Transient temperatures are numerically calculated inside fuel, clad and coolant simultaneously from initial operating conditions, during shutdown and after pumps switching off. Calculations are in good agreement with in-pile data at scram and normal shutdown.

Fuel and clad temperatures rise about 10°C after pumps switching off in hottest bundle.

Fuel temperature does not increase considerably when reactor tripping takes place within 10 % flow reduction irrespective of flow coast down time.

NOMENCLATURE

A	power ratio;
h	heat transfer coeff., W/m^2°C;
K	thermal conductivity, W/m sec.;
per	period; sec.
q	volumetric heat generation, W/m^3;
r	radial coordinate, m;
T	temperature, C;
t	time, sec.;
Δt	time increment, sec.
Δr	space step, m.
α	thermal diffusivity, m^2/sec.;

Subscripts

f	fuel
c	clad
s	coolant

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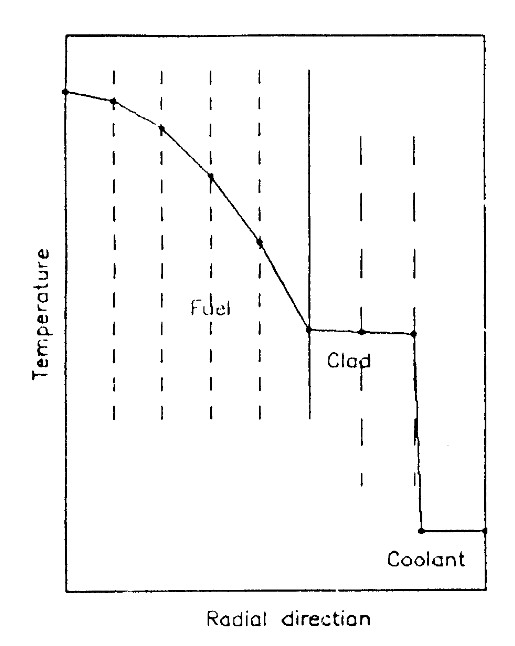


Fig. 1. Nodes in fuel and clad

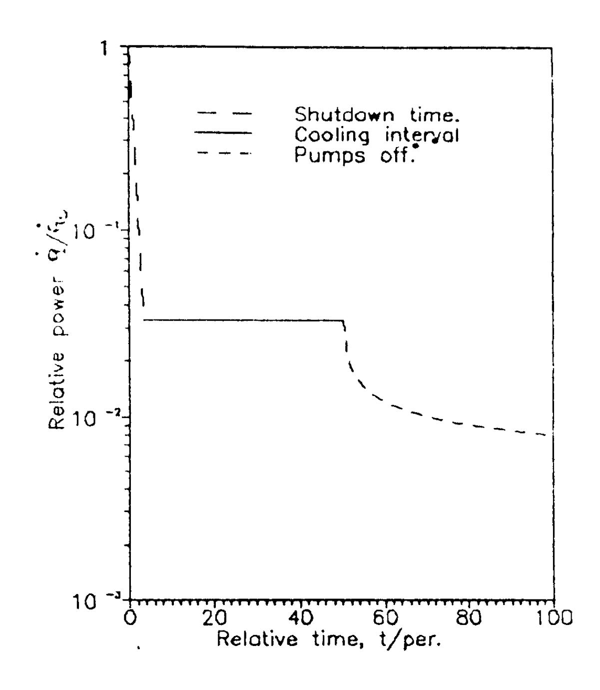


Fig. 2. Relative power during and after shutdown.

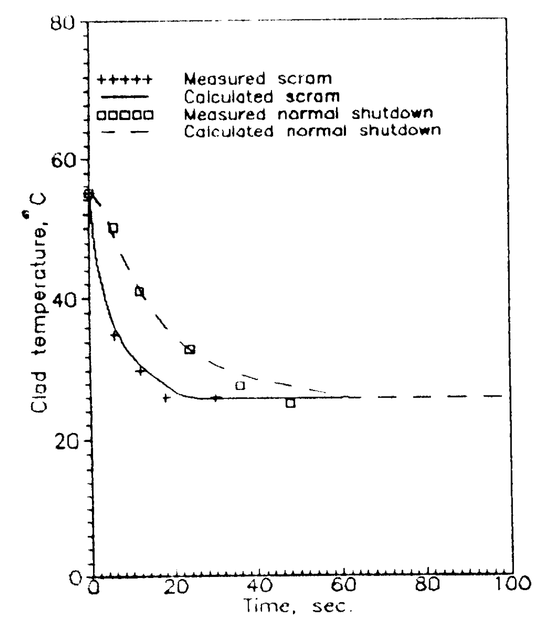


Fig. 3. Transient temperature at different shutdown

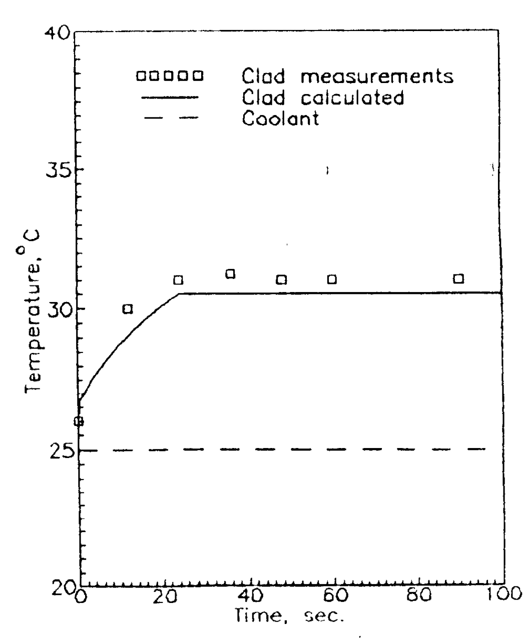


Fig. 4. Temperatures after pumps off.

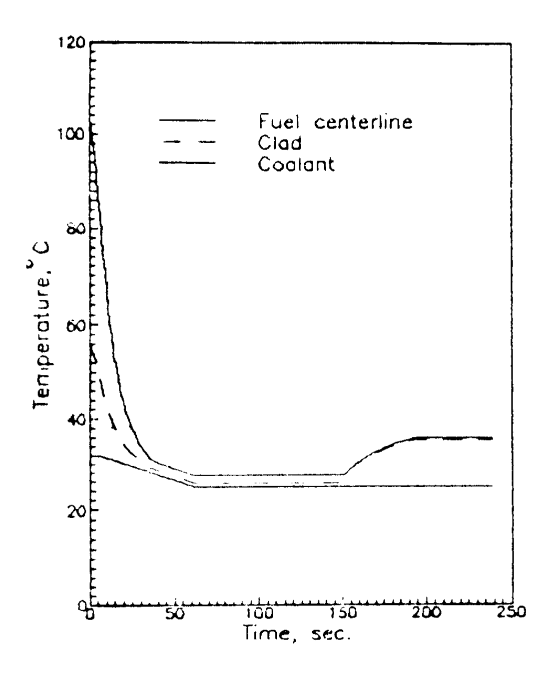


Fig. 5. Transient temperatures at the hottest bundle

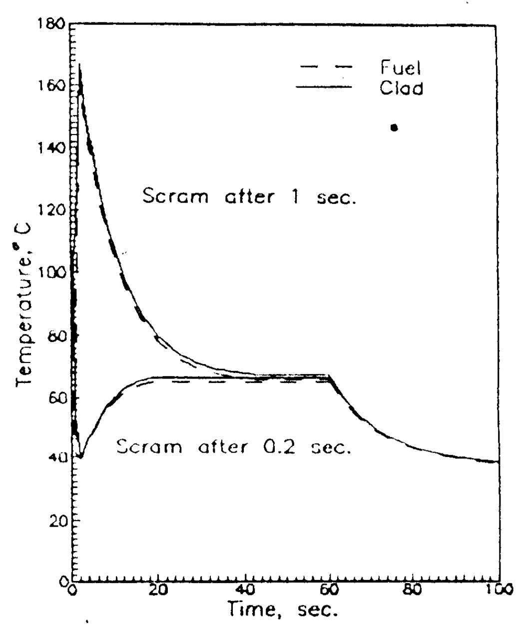


Fig. 6. Effect of scram delay during flow coast down in 2 sec.on temperatures

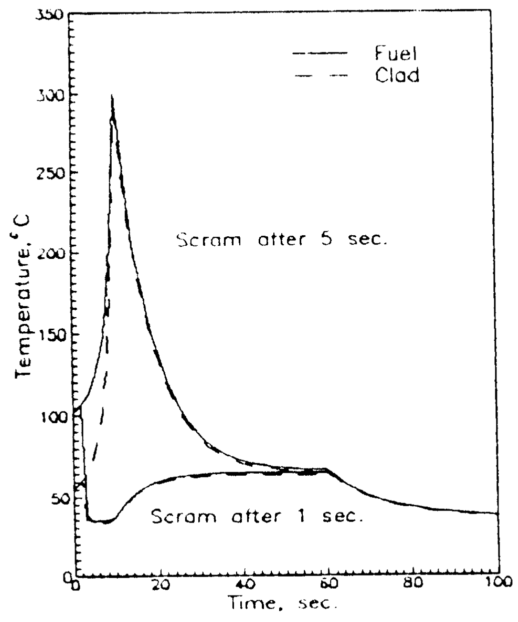


Fig. 7. Effect of scram delay during flow coast down in 10 sec. on temperatures

Lay. 4.2

4.2

Heat Generation

in

Nuclear Reactors

By

A.M. Abdullah

Reactor Heat Generation

1. Heat Generation Rate in Fuel

The rate of nuclear heat generation is equal to the rate of fission reactions times the energy produced per fission.

The energy generated in a reaction per unit time and volume is called volumetric themal source strength, q"' and is given by

$$q''' = G_f \sum_f \phi \qquad (1)$$

G_f = energy generated per fision MeV
$\sum_f$ = avg. mac. fission C.S
ϕ = neutron flux
since $\sum_f = N_{ff} \sigma_f$, hence

$$q''' = G_f N_{ff} \overline{\sigma}_f \phi \quad \text{MeV/cm}^3\text{sec} \qquad (2)$$

where N_{ff} = density of fissionable fuel (U^{235}, Pu^{239}, U^{233}) in nuclei/cm^3.

$$q''' = 1.5477 \times 10^{-8} G_f N_{ff} \overline{\sigma}_f \phi \quad \text{Btu/hr ft}^3$$

The neutron flux distribution in the core is usually perturbed due to fuel, moderator and other materials in a heterogeneous core, the presence of control rods, reflector, changes in fuel enrichment, etc. In much of the discussion that follows we will use as examples the thermal bare homog. Core flux distribution (sinusoidal in the axial direction and Bessel fn in the radial direction).

2. Fission Energy Distribution

The avg. total energy produced perfission = 193 Mev for U^{233},U^{235},Pu^{239}. The energy consists of an instantaneous part (80.5% K.E. of f.p + 2.5% neutrons + 2.5 % γ) and delayed part (0.02% delayd n,s + 3% β-decay of f.p.+ 5% Neutrinos associated with β and nonrecoverable + 3% γ of f.p.).

The delayed part continue to generate heat some time after shutdown. There is in addition about 7 Mev energy generated from the non-fission reaction (n,γ) of the excess n's in the fuel structure, mod., coolant, cladding, etc.

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n,s = neutrons , f.p. = fission products,
K.E = Kinetic Energy

Fission fragments are slowed down in avery short distance (less than 0.01 inch) and their K.E. can be considered to be converted into heat in the fuel at the point of fission. Fission n's both prompt and delayed, are slowed down gradually and in successive scattering processes (in thermal reactor). Their range is medium (from a fraction of an inch to a few feet) and their K.E is converted into heat in the various reactor materials. Gamma energy has long range. A large portion of it escapes the reactor core and is absorbed by the reactor vessel shielding material.

Neutrinos do not react with reactor materials, and the energy carried by them, ~ 5% of the total, is unrecoverable. The remainder is then 95% is long-range. Accordingly about 90% of the total energy per fission is produced in the fuel itself, 4 % in the moderator, 5% by neutrino & 1% in other reactor components. The amount of energy produced in fuel per fission is then 0.90x200 = 180 MeV.

It should be noted that, in new core loading, the amount of energy produced per fission is lower than that given above because the heat given off by the decaying fission products is nonexistent.

After shutdown the decay energy continues to be produced by the still radioactive but decaying f.p. although fission energy completely stops a very short time after reactor shutdown. The decay energy is initially some 3 to 4% of the energy before shutdown. Cooling of the reactor after shutdown is, therefore, necessary.

3. Fissionable Fuel Density

Recall that fuel is defined as all U, Pu, and Th isotopes only, and that fuel material is defined as the entire fuel-bearing material including chemical (such as UO_2 or UO_2SO_4), alloy or mixture (such as U + Al or U + ZrH). The term fuel element refers to fuel material and cladding and other structural materials in the fuel.

The no. of fissionable fuel atoms per cm^3 N_{ff} in a heterog. reactor is usually fixed for any one new fuel element, for an entire new core, or for each of a number (sually three) of concentric zones in a new core loading. These zones are chosen so that they contain nearly equal nos. of fuel elements and are more enriched the farther away from core center. This helps flatten the radial neutron-flux and energy distribn. During core life, N_{ff} will vary because of fuel burnup, cycling (replacing a more burned-up inner zone by a less burned-up outer

zone), during a fuel change, or because of spiking (scattering new elements in the core).

$$N_{ff} = \frac{A_v}{M_{ff}} \rho_{ff}\, i \qquad (3)$$

A_v = Avogadro no , 0.60225 x 10^{24} moleculs/g_m mole
M_{ff} = molecular mass of fissionable fuel
ρ = density of fissionable fuel gm/cm^3
i = no of fuel atoms per molecule of fuel

ρ_{ff} is usually unknown. It is ρ_f, the density of the fuel (all U, Pu and Th isotopes) or ρ_{fm}, the density of the fuel material (all isotopes above plus chemical or alloy compounds, but not cladding) that are known. ρ_{ff} is related to ρ_f by

$$\rho_{ff} = r\, \rho_f = rf\, \rho_{fm} \qquad (4)$$

r = fissionable material mass/fuel mass

$$f = \frac{\text{fuel mass}}{\text{fuel material mass}} = \frac{\mu_f\, M_f}{\sum_i \mu_i M_i}$$

μ_f = no of moles of fuel in the fuel material
M_f = molecular weight of fuel
Usually the fuel is composed of U^{235} & U238 and, hence. M_f is then the equivalent molecular weight of the two isotopes

$$\mu_f\, M_f = \mu^{25}\, M^{25} + \mu^{28}\, M^{28} = \text{mass of } U^{235} + \text{mass of } U^{238}$$

μ^{25} = no of moles (or mole fraction) of U^{235} in the fuel
μ^{25} = mole fraction of U^{25} = number of moles U^{25}/total no of moles of fuel

$$\therefore\ \mu^{25} = \frac{G^{25}/M^{25}}{\frac{G^{25}}{M^{25}} + \frac{G^{28}}{M^{28}}}$$

where G^{25} and G^{28} are the mass fractions of U^{235} & U^{238} in the fuel respectively

$$G^{25} = r\, G\ ,\ G^{28} = (\ 1 - r\ )\ G$$

where G = mass of fuel and r is the % enrichment of U^{235}

$$\therefore\ \mu^{25} = \frac{r/M^{25}}{\frac{r}{M^{25}} + \frac{1-r}{M^{28}}}\ ,\qquad \mu^{28} = \frac{(1-r)/m^{28}}{\frac{r}{M^{25}} + \frac{1-r}{M^{28}}}$$

If the fuel is UO_2

$$\therefore \sum_i \mu_i M_i = \mu_f M_f + 2M_o$$

$$\therefore f = \frac{\mu_f M_f}{\mu_f M_f + 2M_o}$$

r = enrichment or mass ratio of fissionable fuel to total fuel
f = mass fraction in the fuel material. It is then given by:

$$f = \frac{\mu_{ff} M_{ff} + \mu_{nf} M_{nf}}{\mu_{ff} M_{ff} + \mu_{nf} M_{nf} + M_{o_2}} \quad (5)$$

where M_{nf} = molecular mass of nonfiss. fuel (such as U^{238})

M_{o_2} = molecular mass of nonfuel material chemically combined with fuel (such as O_2 in UO_2). Since $M_{ff} \sim M_{nf}$, Eq (5) reduces to

$$f = \frac{\gamma M_{ff} + (1 - r) M_{nf}}{\gamma M_{ff} + (1 - r) M_{nf} + M_{o_2}} \quad (6)$$

combining Eqs. (3) & (4)

$$N_{ff} = \frac{A_v}{M_{ff}} r \rho_{fm} fi \quad (6')$$

Example 1 Calculate N_{ff} for 3% enriched UO_2 for UO_2, ρ_{fm} = 10.5 gm/cm^3, i = 1
M_{ff} = 235.05 , M_{nf} = 238.05

$$\therefore f = \frac{0.03 \times 235.05 + 0.97 \times 238.05}{0.03 \times 235.05 + 0.97 \times 238.05 + 2 \times 16} = 0.8815 \text{ , (6)}$$

$$N_{ff} = \frac{0.60225 \times 10^{24}}{235.05} \times 0.03 \times 10.5 \times 0.8815 \times 1$$

$$= 7.115 \times 10^{20} \; U^{235} \text{ nuclei/cm}^3$$

Fission Cross Section

Thermal Reactors

In thermal reactors, the thermal flux follows the Maxellian distribn, if the absorbing medium is weak. Eq. (2) can be rewritten as

$$q'''(E)\, dE \quad G_f N_{ff} \sigma_f (E) \phi(E) dE \quad (7)$$

$q'''(E)$ ≡ Vol. thermal source strength per unit energy interval dE

$\sigma_f(E)$ = fiss. C.S. at neutron energy E

$\phi(E)$ = neutron flux per unit energy

$$q''' = \int_0^\infty q'''(E)\,dE = G_f\, N_{ff} \int_0^\infty \sigma_f\,(E)\,\phi\,(E)\,dE \tag{8}$$

The avg. C.S can be expressed as

$$\bar{\sigma}_f = \frac{\int_0^\infty \sigma_f\,(E)\,\phi\,(E)\,dE}{\int_0^\infty \phi\,(E)\,dE} \tag{9}$$

Eq. (8) becomes

$$q''' = G_f\, N_{ff}\, \bar{\sigma}_f \int_0^\infty \phi(E)\,dE \tag{10}$$

Also,

$$\sigma_f\,(E) = \sigma_{fo}\sqrt{\frac{E_o}{E}} \qquad \text{for } \frac{1}{V} \text{ law} \tag{11}$$

where

E_o = 0.0253 ev (thermal neutron energy at 20 °C)

σ_{fo} = fiss. C.S at E_o

Eq. (9) becomes

$$\bar{\sigma}_f = \sigma_{fo}\sqrt{E_o}\,\frac{\int_0^\infty E^{-1/2}\,\phi(E)\,dE}{\int_0^\infty \phi\,(E)\,dE} \tag{12}$$

Assume the thermal neutron density obeys the Maxwellian distribution as

$$n(E)\,dE = \frac{2\,\pi\,n}{(\pi\,K\,T)^{3/2}}\sqrt{E}\,e^{-E/KT}\,dE \tag{13}$$

n(E) = no of n's per unit volume per unit energy

n = total no of n's

K = Boltzmann's const.

T = abs. temp

Note that $E = 1/2\, m\, V^2$ & $n(E)\,dE = n(V)\,dV$ where m and V are mass and velocity of the neutron.

substitute for $n(E) = \frac{\Phi(E)}{V} = E^{-1/2}\,\phi(E)\sqrt{\frac{m}{2}}$

Eq. (13) becomes

$$\phi(E)\,dE = \sqrt{\frac{2}{m}}\,\frac{2\pi n}{(\pi KT)^{3/2}}\,E\,e^{-E/KT}\,dE$$

substitute in (12)

$$\bar{\sigma}_f = \sigma_{fo}\sqrt{E_o}\,\frac{\int_0^\infty \sqrt{E}\,e^{-E/KT}\,dE}{\int_0^\infty E\,e^{-E/KT}\,dE}$$

$$= \sigma_{fo}\sqrt{E_o}\,\frac{\frac{1}{2}KT(\pi KT)^{1/2}}{(KT)^2}$$

or

$$\bar{\sigma}_f = \frac{\sqrt{\pi}}{2}\,\delta_{fo}\sqrt{\frac{E_o}{KT}}$$

where E_o is the energy corresponding to the most probable speed at temp. To Hence $E_o = KT_o$, consequently

$$\bar{\sigma}_f = 0.8862\,\sigma_{fo}\sqrt{\frac{T_o}{T}} \tag{14}$$

where σ_{fo} = fission cross section at 0.025 eV or 2200 m/sec

T_o=20° +273 = 293°K =(32 +9/5 x 20)°F+460 = 528 °R

T is the effective neutron temp., , it is that of the fuel-moderator mixture in a homogeneous reactor. Since the neutrons are slowed down in the moderator, where the scatt cross section is much greater than the absorption cross section, we can take, approximately, the neutron temp. to be equal to that of the moderator in heterogeneous reactors.

The distribution (13) is applied between E = 0 & E ~ 5 KT where T is the neutron temp. Above 5KT the flux (epithermal flux) varies as 1/E and extended to same energy E_p. Above E_p the fission neutron spectrum starts which can be represented by the semi-empirical equation

$$n(E) = \sqrt{\frac{2}{\pi e}}\,e^{-E}\,\sinh\sqrt{2E} \tag{15}$$

In any case, the true average cross section should be based on a modified neutron-flux distribution $\phi'(E)$ =

$\phi(E) + \phi_e(E)$, where $\phi(E)$ is the Maxwellian distribution and $\phi_e(E)$ is the epithermal flux.

At energies above the thermal energy, the fission cross section does not follow the 1/V law, and this can be corrected by introducing the factor $g_f(T)$, called the non-1/V factor, hence for Maxwellian distribution

$$\bar{\sigma}_f = 0.8862\, g_f\,(T)\, \sigma_{fc} \sqrt{\frac{T_o}{T}} \qquad (16)$$

where T is the effective neutron temperature (~ moderator temp.) .
The non-1/V absorption is found only in intermediate and heavy nuclei. The factor $g_f(T)$ was introduced by Westcott.

Example 2 . Calculate the thermal source strength corresponding to a neutron flux of 10^{13} neutrons/cm^2.sec for UO_2 with 3% enrichment. The moderator temp. is 500°F, and the fuel material density is 10.5 gm/cm^3

$$N_{ff} = \frac{A_v}{M_{ff}}\, rf\, \rho_{fm}$$

$$\frac{0.6023 \times 10^{24} \times 0.03 \times 0.8815}{235} \times 10.5 = 7.11 \times 10^{20}\ U^{235} \text{ atom}$$

where f has been calculated from (6).

$\sigma_{fo} = 577$ barns, $g_f\ (T) = 0.93$ at $T = 500°F$

from (16) ,

$$\bar{\sigma}_f = 0.8862 \times 0.93 \times 577 \sqrt{\frac{528}{960}} = 352.7 \text{ barn}$$

consider G_f = 180 Mev / fission

$$\therefore q_f''' = G_f \textstyle\sum_f \phi = 180 \times 352.7 \times 10^{-24} \times 7.11 \times 10^{20} \times 10^{13}$$
$$= 4.514 \times 10^{14} \text{ Mev/cm}^3 \text{ . sec}$$
$$= 4.514 \times 10^{14} \times 1.5477 \times 10^{-8} = 6.986 \times 10^{6} \text{ Btu/hr.ft}^3$$

Fast Reactors

In fast reactors, the fission cross sections for U^{235}, Pu^{239}, and U^{233} are approximately constant with the neutron energy up to about 10^5 ev for U^{235} and U^{233} , and

5×10^3 eV for Pu^{239}. Below these energies the fission cross sections increase as $1/E^n$. Above these energies they may be taken as 1.25, 1.8, and 2 barns, on the average, for the above three materials respectively. For the fertile fuels Pu^{240}, U^{238}, and Th^{232} the fission cross sections may be taken approximately as 1.6, 0.6, and 0.2 barns (on the average) respectively. For energies below 1 Mev the fission cross sections of U^{238} and Th^{232} drop to zero. When used in fast reactor, U^{238} contributes between 10 and 20% of the total fission energy.

Heat Generated in a Fuel Element

From reactor physics, the ideal neutron flux was calculated as a pure cosine distribution in the Z - direction. This represents, actually, the case of a bare homogeneous reactor. Under certain conditions & approximations, this cosine distribution can be considered to be applicable to a heterogeneous reactor.

However, if the reactivity varies with Z, because of a large axial temp. rise in a water-moderated reactor or because of change in phase where the moderator density decreases such as in BWR's, or a change in fuel enrichment due to unequal burnup or because of partially inserted control rods, the axial flux may deviate strongly from the pure cosine distribution.

The total heat generated in a fuel element, q_t, is

$$q_t = \int_{-H/2}^{H/2} q'''(Z)\, A_c\, dZ \qquad \text{Btu/hr}$$

where

$q'''(z)$ = volumetric thermal source strength

A_c = fuel metal cross sectional area

If $q'''(Z) = q'''_c \cos(\pi Z/H_e)$, then

$$q_t = q_c A_c \int_{-H/2}^{H/2} \cos\left(\frac{\pi Z}{H_e}\right) dZ$$

or

$$q_t = \frac{2}{\pi} q_c A_c He \sin\left(\frac{\pi H}{2 H_e}\right) \qquad (17)$$

where

q'''_c = volumetric thermal source strength at the fuel mid point

H_e = extrapolated height of the core or fuel element

Total Heat Generated in Reactor Core

In a heterogeneous core the neutron flux ϕ_c and hence q''' vary in the radial direction. They are max. at the core center and min. at the boundaries. In order to treat the heterogeneous core as a homogeneous one, two conditions must be satisfied: (1) the number of fuel elements in the heterogeneous core should be very large, in this the cross sections of the elements are small and consequently the variation in the radial flux within any element can be neglected, and (2) the fuel type and enrichment should not vary in the heterogeneous core, or they vary in a manner that can be treated analytically or numerically.

Heat Generated in a Homogeneous Core

In this core, the volumetric thermal source strength is expressed as

$$q'''(r) = G_f N_{ff} \overline{\sigma}_f \phi(r)$$

The value of G_f in a homogeneous core includes the heat generated in the moderator (due to neutron slowing down and radiation absorption) which averages about 5% of the total heat generated. In a homogenous core G_f = 190 MeV/fission, and 180 in a heterogeneous one.

Example 3. Determine the heat generated, Q_t (Btu/hr) in a bare homogeneous spherical reactor of radius R(ft). Neglect the difference between the extrapolated and actual radii.

Solution

From reactor physics

$$\phi(r) = \phi_{co} \frac{\sin(\pi r/R)}{\pi r/R} \qquad (18)$$

Hence

$$q'''(r) = G_f \sum_f \phi(r) = G_f N_{ff} \overline{\sigma}_f \phi_{co} \frac{\sin(\pi r/R)}{\pi r/R}$$

Consider an element of volume $4\pi r^2$ dr

$$Q_t = \int_0^R q'''(r)\, 4\pi r^2 \, dr$$

$$= \frac{4\pi G_f N_{ff} \overline{\sigma}_f \phi_{co}}{\pi/R} \int_0^R r \sin\left(\frac{\pi r}{R}\right) dr \quad \text{Mev/Cm}^3 \cdot \text{sec}$$

since 1 Mev/cm^3 sec = 1.5477×10^{-8} Btu/ft^3 hr, hence

$$Q_t = 1.176 \times 10^{-5} N_{ff} \bar{\sigma}_f R \phi_{co} \int_o^R r \sin (\frac{\pi r}{R}) dr$$

$$= 1.176 \times 10^{-5} N_{ff} \bar{\sigma}_f R \phi_{co} \left[(\frac{r}{\pi})^2 \sin\frac{\pi r}{R} - \frac{R}{\pi} r \cos \frac{\pi r}{R} \right]_o^R$$

$$= 3.745 \times 10^{-4} N_{ff} \bar{\sigma}_f \phi_{co} R^3 \qquad (19)$$

Heat Generation in a Heterogeneous Reactor Containing Large Number of Fuel Elements

In this case, which is a characteristic of power reactors, Q_t can be calculated, with little error, by considering the heterogenous core as a homogeneous one and modifiying q"' of the homogeneous core as

$$q'''_{het} = q''' \frac{\text{fuel volume}}{\text{core volume}} = q''' \frac{V_{fuel}}{V_{core}} \qquad (20)$$

where q'''_{het} is the thermal source strength of the homogenized heterogeneous core. An alternate procedure can be used as demonstrated in the following example

Example 4. compute Q_t (Btu/hr) for a cylindrical core of radius R, height H containing n vertical fuel elements.
Solution:
The flux distribution in a bare homogeneous cylindrical core is

$$\phi(r,z) = \phi_{co} \cos (\frac{\pi z}{H_e}) J_o (\frac{2.405 r}{R_e}) \qquad (21)$$

compare this eq. with

$$\phi_{(z)} = \phi_c \cos (\frac{\pi z}{H_e}) \qquad (22)$$

we get

$$\phi_c = \phi_{co} J_o (\frac{2.405 r}{R_e}) \qquad (23)$$

where ϕ_{co} is the flux at the reactor center
Hence

$$q_c''' = q_{co}''' J_o (\frac{2.405 r}{R_e}) \qquad (24)$$

Now, let q_t"' be the average heat generated per unit cross sectional area in a heterogeneous core, then

$$q_t'' = \frac{n q_t}{\pi R^2} \qquad (25)$$

where q_t is the heat generated from one fuel element.

Using (17) neglecting the extrapolated distance

$$q_t'' = \frac{2\,n}{\pi^2 R^2} A_c\, H\, q_c''' \quad (26)$$

Using (24)

$$q_t''(r) = \frac{2\,n}{\pi^2 R^2} A_c\, H\, q_{co}'''\, J_o\left(\frac{2.405\, r}{R_e}\right) \quad (27)$$

$$\therefore\; Q_t = \int_0^R q_t''(r)\, 2\pi\, r\, dr = \frac{4n}{\pi R^2} A_c\, H\, q_{co}''' \int_0^R r\, J_o\left(\frac{2.405\, r}{R}\right) dr$$

$$= \frac{4n}{\phi\, R^2} A_c H\, q_{co}''' \frac{R_e}{2.405} \left[r\, J_1\left(\frac{2.405\, r}{R_e}\right)\right]_o^R$$

Considering $R_e = R$, and using $J_1\,(2.405) = 0.519$, hence

$$Q_t = \frac{4n}{\pi R^2} A_c H\, q_{co}'''\, (0.2158R^2)$$

or

$$Q_t = 0.275n\, A_c H\, q_{co}''' \quad (28)$$

This is the heat generated in the solid fuel alones. To account for the heat generated in the moderator and other reactor materials we multiply the R.H.S. of (28) by 1.05, hence

$$Q_t = 0.2888\, (nA_c\, H)\, q_{co}''' \quad (29)$$

In the case of a small number of relatively large fuel elements such as in research or training reactors, the total heat generated in the core can be found by calculating and summing the heat generated in the individual fuel elements.

Reactor Shutdown Heat Generation

In reactor shutdown, the reactor power falls off rapidly according to a negative period, which determined by the half-life of the longest-lived delayed neutron group, i.e the group which has the smallest decay constant. Also, the fission fragments and fission products existing in the fuel continue to decay, β and γ, at decreasing rates, for long periods. Therefore, the reactor continues to generate power P_s after shutdown. The magnitude of the power P_s depends on :
(a) the reactor power level P_o before shutdown
(b) the length of time, θ_o, the reactor has operated at the power level P_o; since both these factors determine the amount of fission products present.

Due to this decay heat, it is essential to cool the

reactor after shutdown, otherwise the fuel and cladding temps. will rise and may cause structural damage of fuel elements and release of radioactivity. The ratio of the volumetric thermal source strength after shutdown q_s''' to that before shutdown q_o''' could be expressed as

$$\frac{q_s}{q_o} = \frac{P_s}{P_o} \tag{30}$$

For uranium fuels, the following empirical equation was obtained

$$\frac{P_s}{P_o} = 0.095\ \theta_s^{-0.26} \tag{31}$$

where $\theta_s > 200$ sec. and $\theta = \infty$ (i.e greater than 1 year).
For $\theta_o < 1$ year, we use the equation:

$$\frac{P_s}{P_o} = 0.095\ \theta_s^{-0.26} \left[1 - \left(1 + \frac{\theta_o}{\theta_s} \right)^{-0.2} \right] \tag{32}$$

The total energy released after shutdown E_s is

$$E_s = \int_0^{\theta_s} P_s\, d\theta_s = 0.128\ \theta_s^{0.74}\ P_o \tag{33}$$

Example 5. A fast reactor uses fuel material of density 10 gm/cm^3 and composed of 80% depleted UO_2 and 20% PuO_2 by mass. At the core center the flux is 10^{16} and the effective fission cross section of Pu^{239} is 1.78 borns. Ignoring the content of U^{235} in the fuel and assuming that plutonium is 80 % enriched in fissionable Pu^{239} (the rest being nonfissionable Pu^{240}) calculate q_s''', E_s one hour after shutdown assuming the reactor has operated at the above flux for more than one year, also calculate q_s''' after one hour of shutdown if the reactor has operated 6 weeks at the same flux.
Solution The mass fraction of Pu in the Pu O_2 fuel, using (6), is

$$f_{Pu} = \frac{0.8 \times 239.05 + 0.2 \times 240.05}{0.8 \times 239.05 + 0.2 \times 240.05 + 2 \times 16} = 0.882$$

The mass fraction of Pu O_2 in the total fuel is 0.2, hence the mass fraction of Pu^{239} in the total fuel is 0.2 x 0.882 = 0.1764, this is equal to the quantity rf in Eq (4) .

$N_{ff} = i \dfrac{\rho_{fm} A_v}{M_{ff}}$ x mass fraction of fissionable fuel in the

fuel material,

hence,

$$N_{11} = 1 \times \frac{10 \times 0.6023 \times 10^{24}}{239.05} \times 0.1764 = 4.4445 \times 10^{21} \text{ atoms/cm}^3$$

The volumetric thermal source strength at the reactor center is

$$q_o = G_f \sum_f \varphi_o = 180 \times 1.78\times10^{-24} \times 4.4445\times10^{21} \times 10^{16} = 1.424\times10^{16} \frac{\text{Mev}}{\text{cm}^3.\text{sec}}$$

$$1.424\ 10^{16} \times 1.5477 \times 10^{-8} = 2.204 \times 10^{8} \text{ Btu/hr.ft}^3$$

$$\frac{q_s}{q_o} = \frac{P_s}{P_o} = 0.095(\theta_s)^{-0.26} \text{ where } \theta_s \text{ in sec}$$

$$\therefore\ q_s = 0.095(3600)^{-0.26}\, q_o = 2.491 \times 10^{6} \quad \text{Btu/hr ft}^3$$

$$E_s = 0.128\ \theta_s^{0.74}\ P_o = 0.128(3600)^{0.74}\ P_o = 54.81 \text{ sec per unit power}$$

$$\frac{54.81}{3600} = 1.523 \text{ Btu/(Btu/hr)} = 1.523 \frac{\text{Btu/ft}^3}{\text{Btu/hr ft}^3}$$

Then after one hour from shutdown, the energy released is

$$E_s = 1.523 \frac{\text{Btu/ft}^3}{\text{Btu/hr ft}^3} \times 2.491 \times 10^{6} \frac{\text{Btu}}{\text{hr. ft}^3} = 3.794 \times 10^{6} \text{ Btu/ft}^3$$

For operating time less than one year before shutdown, we use Eq. (32).

$$q_s = 2.491 \times 10^{6} \times \left[1 - \left(1 + \frac{6 \times 7}{1/24}\right)^{-0.2}\right]$$

$$= 2.491 \times 10^{6}\ (1 - 0.2507) = 1.866 \times 10^{6} \quad \text{Btu/ft}^3.\text{ hr}$$

HEAT REMOVAL FROM NUCLER REACTORS

A.M. ABDULLAH

Basic heat transfer Eqs.

$$Q = hA(t_c - t_f)$$

Definitions of t_c, t_f, A, h

$$h = C \frac{K}{D_e} Re^{0.8} Pr^{0.4}, \quad Re > 10^4 \ (\text{Turb.})$$

$$C = 0.042\left(\frac{P}{D}\right) - 0.024$$

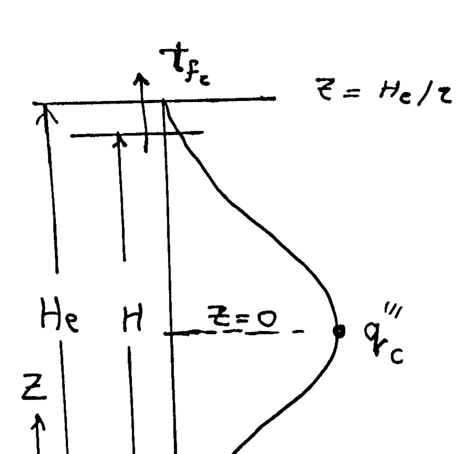
$$Re = \frac{\rho u D_e}{\mu}, \quad Pr = \frac{\mu C_p}{K}$$

$$D_e = 4 \frac{\text{flow area}}{\text{Wetted perimeter}}$$

Axial Temp. Distb. of Reactor Coolant

Assume the neutron flux varies as

$$\phi = \phi_c \cos\left(\frac{\pi z}{H_e}\right)$$



For bare core

$$H_e = H + 1.42 \lambda_{tr}$$

$$\lambda_{tr} = 1/\Sigma_s(1-\bar{\mu}_0), \quad \bar{\mu}_0 = \frac{2}{3A}$$

(A = mass. no. of scatt. nuclei)

For reflected core

$$H_e = H + 2\delta$$

δ = reflector saving

$$\delta = \frac{D_c}{D_r} L_r \tanh(T/L_r), \quad T \equiv \text{reflect. thick.}$$

Power density $q_c''' = \gamma\, G\, \bar{\Sigma}_f\, \phi_c$ W/cm³

$$q''' = q_c''' \cos\left(\frac{\pi z}{H_e}\right)$$

Heat balance across a small length dz is.

$$\dot{m}\, C_p\, dt_f = q''' A_c\, dz \qquad (\text{Defin. } \dot{m},\ t_f,\ A_c)$$

or

$$\dot{m}\, C_p\, dt_f = q_c''' A_c \cos\frac{\pi z}{H_e}\, dz$$

Integrate from $z = -H/2$ to $z = z$, and from $t_f = t_{f_1}$ to $t_f = t_f$

$\therefore$

$$t_f = t_{f_1} + \frac{q_c''' A_c H_e}{\pi \dot{m} C_p}\left(\sin\frac{\pi z}{H_e} + \sin\frac{\pi H}{2H_e}\right)$$

Axial Temp. Distrib. of fuel Element

At st. st. condition the heat generated within the fuel must be transferred to the coolant through the cladding

$$h\, C\, dz\, (t_c - t_f) = q_c''' \cos\left(\frac{\pi z}{H_e}\right) A_c\, dz$$

from which

$$t_c = t_f + \frac{q_c''' A_c}{h C} \cos\left(\frac{\pi z}{H_e}\right)$$

$C \equiv$ circumferencial length of clad surface. For cylind. fuel

$$C = 2\pi(R + \delta_g + \delta_c)$$

R = fuel radius, δ_g = gas gap thick., δ_c = clad thick.

$$t_c = t_{f_1} + q_c''' A_c\left[\frac{1}{hC}\cos\frac{\pi z}{H_e} + \frac{H_e}{\pi \dot{m} C_p}\left(\sin\frac{\pi z}{H_e} + \sin\frac{\pi H}{2H_e}\right)\right]$$

The max. clad surface temp. can be calculated from the condition:

$$\left.\frac{dt_c}{dz}\right|_{z=z_c} = 0$$

from which

$$z_c = \frac{H_e}{\pi} \tan^{-1}\left(hC H_e / \pi \dot{m} C_p\right)$$

Temp. Drop Across the Clad, Gas Gap and Fuel

Clad

Heat transfer in the fuel is transferred through the Clad or

$$q'''(\pi R^2 dz) = K_c\left(\frac{t_s - t_c}{\delta_c}\right) 2\pi (R + \delta_g)\, dz$$

$\therefore$

$$t_s = t_c + \frac{q_c''' R^2}{2K_c}\left(\frac{\delta_c}{R+\delta_g}\right) \cos\frac{\pi z}{H_e}$$

where t_s is the clad inside surface temp

Gas Gap, as before

$$q'''(\pi R^2 dz) = h_g (2\pi R\, dz)(t_{Fs} - t_s)$$

from which

$$t_{Fs} = t_s + \frac{R q_c'''}{2h_g} \cos\left(\frac{\pi z}{H_e}\right)$$

where t_{Fs} is the fuel surface temp.

Fuel

$$q'''(\pi R^2 dz) = K_f\left(\frac{t_m - t_{Fs}}{R/2}\right) 2\pi R\, dz$$

from which

$$t_m = t_{Fs} + \frac{q_c''' R^2}{4K_f} \cos\left(\frac{\pi z}{H_e}\right)$$

From the above Eqs:

$dt_s/dz = 0$, $dt_{Fs}/dz = 0$, $dt_m/dz = 0$, can be calculated

For fuel elements made of uranium metal the max. temp. t_{mm} should be kept below 1224°F at which phase change from α to β phase occurs in the structure of the material. Due to this phase change the material grows and may cause the fuel element to buckle. This is undesirable for it may cause obstruction of the coolant flow passages and subsequent burn-out of the fuel elements. On the other hand, uranium dioxide UO_2 presents no problems regarding phase changes, its melting point is very high (~5000°F). However, due to its low thermal conductivity the temp. drop within it becomes relatively high which results in high thermal stresses within the fuel element.

Example : The conditions at the cross-section of max. surface temp. in a clad plate fuel element 3.5 in wide, 0.2 in thick, and 4 ft long, clad in 0.005 in. thick 304L stainless steel are given as

$$\phi = 3 \times 10^{13} \text{ neutrons}/cm^2 \text{ sec}$$
$$q''' = 2.185 \times 10^7 \text{ Btu}/ft^3 \text{ hr}$$
$$h = 3480 \text{ Btu}/hr\ ft^2\ °F$$
$$t_m = 700°F, \quad t_s = 658.7°F, \quad t_c = 652.3°F \ \&\ t_f = 600°F$$

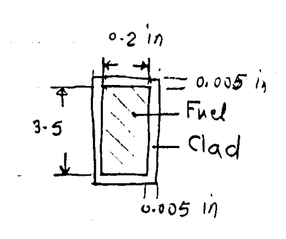
Requirements: ϕ_c, q'''_c, q_t, t_{f_1}, t_{f_2}, & pressure

If light water acts as coolant-moderator and infinite reflector (in the axial direction) and flows at a rate of 3 lbm/sec per fuel element, calculate (a) the max. neutron flux occurring in the fuel element, (b) the max. volumetric thermal source strength, (c) the total heat generated by the fuel element, (d) the inlet and outlet temps. of the coolant, (e) the min. coolant pressure necessary to avoid local boiling of the coolant.

(f) Assume now, the system is operating under const. press. equals to that calculated in requirment (e), determine the max. clad surface temp. and outlet coolant temp. if the mass flow has decreased by 50% due to pump failure (LOFA). Consider the inlet coolant temp. to be the same as that calculated before, $[h' = h(\frac{1}{2})^{0.8}]$. Is there local boiling on the clad surface?

$$\phi_c = \phi / \cos(\pi Z_c / H_e)$$

neglect the gap thickness, from (15)



$$Z_c = \frac{H_e}{\pi} \tan^{-1}\left(\frac{h C H_e}{\pi \dot{m} C_p}\right)$$

$$C = 2(3.5 + 2 \times 0.005 + 0.2 + 2 \times 0.005)$$
$$= 7.44 \text{ in} = 0.62 \text{ ft}$$

Since the inlet and outlet temps. are still unknown, so we consider the 600°F to be the average temp. at which C_p is calculated as $C_p = 1.45$ Btu/lb$_m$°F.

H_e, the extrapolat height of a bare thermal reactor, is related to the actual height H by . $H_e = H + 2\Delta$, where Δ is the reflector saving. This can be shown to be, for large reators, equal to

$$\Delta \approx \frac{D_c}{D_r} L_r \tanh\left(\frac{T}{L_r}\right) \rightarrow \text{One-group}$$

where D_c & D_r are the diff. coeff. in the core & reflector respectively. L_r is the diff. length in the reflector & T is the reflector thickness. For very thick reflector ($T > 2L_r$), $\tanh(T/L_r) \rightarrow 1$. If the diff. coeffs. in the core & reflector are equal, then $\Delta = L_r$. For H_2O of density 1 gm/cm^3, $L_r = 2.88$ cm $= 0.0945$ ft.

$$H_e = 4 + 2 \times 0.0945 = 4.189 \text{ ft}$$

$$Z_c = \frac{4.189}{\pi} \tan^{-1} \frac{3480 \times 0.62 \times 4.189}{\pi \times 3 \times 3600 \times 1.45} = \frac{4.189}{\pi} \tan^{-1} 0.1837$$

$$= \frac{4.189}{\pi} \times 10.41^\circ \times \frac{\pi}{180^\circ} = 0.242 \text{ ft}$$

(a) $$\phi_c = \frac{3 \times 10^{13}}{\cos\left(\frac{10.41\pi}{180} \times \frac{180}{\pi}\right)} = \frac{3 \times 10^{13}}{0.9835} \approx 3.05 \times 10^{13} \text{ n/cm}^2\text{ sec}$$

(b) $$q'''_c = G \Sigma_f \phi_c = G\left(\frac{r \rho_f A_v}{A^{25}} \bar{\sigma}_{ff}\right) \phi_c \quad \text{Mev/cm}^3\text{.sec}$$

assumptions $\begin{cases} r \approx 0.01, \ \rho_f \approx 18.3 \text{ gm/cm}^3 \\ G = 180 \text{ Mev} \end{cases}$

$$q'''_c = (0.9 \times 200\,\text{Mev})\left(\frac{0.01 \times 18.3 \times 0.6 \times 10^{24}}{235}\,\frac{\times 550 \times 10^{-24}}{\text{Cm}}\right)\left(\frac{3.05 \times 10^{13}}{\text{Cm}^2\,\text{sec}}\right)\underbrace{1.5477 \times 10^{-8}\,\frac{\text{Btu/hr ft}^3}{\text{Mev/sec Cm}^3}}_{\text{conversion factor}}$$

$$q'''_c = 2.183 \times 10^7 \ \text{Btu/hr ft}^3$$

(C) $$q_t = \int q''' \, dV = \int_{-H/2}^{H/2} q'''_c \cos\left(\frac{\pi z}{H_e}\right) A_c \, dz$$

$$= \frac{2}{\pi} q'''_c A_c H_e \sin\left(\frac{\pi H}{2 H_e}\right)$$

$$= \frac{2}{\pi} \times 2.183 \times 10^7 \times \frac{3.5 \times 0.2}{144} \times 4.189 \ \sin\left(\frac{\pi \times 4}{2 \times 4.189} \times \frac{180}{\pi}\right)$$

$$= 2.83 \times 10^5 \sin 86^\circ = 2.83 \times 10^5 \times 0.997 = 2.823 \times 10^5 \ \text{Btu/hr}$$

(d) $$\frac{\pi z_c}{H_e} = \frac{\pi \times 0.242}{4.189} \times \frac{180}{\pi} = 10.4^\circ, \text{ hence from (12)}$$

$$t_{f_1} = t_{cm} - q'''_c A_c \left[\frac{1}{hC} \cos \frac{\pi z_c}{H_e} + \frac{H_e}{\pi \dot{m} c_p}\left(\sin \frac{\pi z_c}{H_e} + \sin \frac{\pi H}{2 H_e}\right)\right]$$

$$= 652.3 - 2.183 \times 10^7 \times \frac{3.5 \times 0.2}{144}\left[\frac{\cos 10.4^\circ}{3480 \times 0.62} + \frac{4.189}{\pi \times 3 \times 3600 \times 1.45}\left(\sin 10.4^\circ + \sin \frac{\pi \times 4}{2 \times 4.189} \times \frac{180}{\pi}\right)\right]$$

$$= 652.3 - 1.06 \times 10^5 \left[4.559 \times 10^{-4} + 0.85 \times 10^{-4}(0.18 + 0.997)\right]$$

$$\therefore t_{f_1} = 652.3 - 58.95 = 593.4\ ^\circ\text{F}$$

The outlet coolant temp. is calculated from the heat balance equation

$$\dot{m} c_p (t_{f_2} - t_{f_1}) = q_t$$

$$\therefore t_{f_2} = t_{f_1} + \frac{q_t}{\dot{m} c_p} = 593.4 + \frac{2.823 \times 10^5}{3 \times 3600 \times 1.45} = 611.4\ ^\circ\text{F}$$

(e) To avoid local boiling, the max. coolant temp., which is equal to the clad surface temp, should be less than the sat. temp. corresponding to the system pressure. The sat. pressure corresponding to max. coolant temp. (652.3) is $\simeq$ 2250 psia. This is the min. pressure required to avoid partial subcooled boiling.

(f) repeat the necessary calculations with $\dot{m} = 1.5$ instead of 3 lb/sec.

2.c

Example. The inlet coolant temp. of a PWR core is 590 °F and the max. neutron flux is 10^{14} neutions / cm^2 sec. The fuel radius is 0.47 in. and its length is 5 ft, aluminum clad thick. of 0.02 in. The clad is isolated from the fuel by a gap containing an inerb gas helium of 0.006 in thickness. The core is reflected by a 4 in. thick H_2O in the axial direction. If the coolant velocity is 15 ft/sec. and its flow area inside the core is 2.417 in^2. determine (a) the volumetric thermal source strength (b) outlet coolant temp. (c) the max. temps. of the inside and outside surfaces of the clad material and those of the surface and centre of the fuel and the corresponding positions of these max. temps. (d) total heat generation rate

- heat generated per fission = 180 Mev
- fuel density = 18.5 gm/cm^3, enrichment 1 percent
- average thermal fission c-s 550 barns
- Thermal conductivities of Helium, clad, and fuel are respectively 0.15, 117 and 18 Btu/hr ft-°F
- The Dittus-Boelter equation for calculating the H-T coeff with the coolant properties evaluated at the sat. pressure corresponding to the core inlet temp.

(a) $q_c''' = G \cdot \Sigma_f \, \phi_c = G \left(\frac{r \rho_f A_v}{A^{25}} \bar{\sigma}_f \right) \phi_c$

$= 150 \left(0.01 \times \frac{18.5 \times 0.6 \times 10^{24}}{235} \times 550 \times 10^{-24} \right) \times 10^{14}$ Mev / cm³ sec

$= 46.76 \times 10^{14}$ Mev / cm³ sec $\times 1.5477 \times 10^{-8} \frac{\text{Btu / hr ft}^3}{\text{Mev / cm}^3 \text{ sec}}$

$= 7.[illegible]7 \times 10^{7}$ Btu / hr ft³

$H_e = H + 2\Delta , \quad \Delta \approx L_r \tanh\left(\frac{T}{L_r}\right) \approx 2.88 \tanh \frac{10.16}{2.88} \approx 2.875$ cm

$\therefore H_e = 5 + 2 \times 2.875 \frac{1}{2.54 \times 12} = 5.188$ ft

$q_t = \int_{-H/2}^{H/2} q_c''' \cos\left(\frac{\pi z}{2 H_e}\right) A_c \, dz = \frac{2 q_c''' A_c H_e}{\pi} \sin\left(\frac{\pi H}{2 H_e}\right)$

$= \frac{2 \times 7.[illegible] \times 10^{7}}{\pi} \times \pi \frac{0.47^2}{144} \times 5.188 \sin\left(\frac{\pi \times 5}{2 \times 5.188} \times \frac{180}{\pi}\right)$

$= 11.5[illegible] \times 10^{5} \times 0.998 = 11.5 \times 10^{5}$ Btu / hr

(b) outlet coolant temp.

$t_{f_2} = t_{f_1} + \frac{2 q_c''' A_c H_e}{\pi \dot{m} c_p} \sin \frac{\pi H}{2 H_e} = t_{f_1} + \frac{q_t}{\dot{m} c_p}$

$\dot{m} = \rho V \times$ flow area $= 42.55 \times 15 \times \frac{2.417}{144} = 10.71$ lbm /sec

At $t = 590°$ F, $C_p \approx$ ⟶ Complete

Reactor Instrumentation (Detectors)
M.k.Shaat

Start-up channel: (Fission chamber)

The purpose of this channel of instrumentation is to provide information on the reactor behavior at low neutron levels when other instruments are relatively insensitive. The neutron detector is a fission chamber. The F.ch. has a U^{235} lining in which the neutrons produce fissions. The highly ionized fission fragments produce large voltage pulses, therefore, discrimination against pulses resulting from the large gamma background present in the reactor and from -porticles from the uranium is relatively easy. the pulses from the F. ch. are preamplified, then shaped and amplified in a linear amplifier, and passes through a discriminator circuit. This outlut is usually converted to a count rate and displayed on a logcount-rote meter on the reactor console. A log count-rate meter is used so that General decodes of counting rote can be displayed on the same meter. For some reactors the logcount-rate signal is poned through a differentiation circuit, and the period is obtained and displayed on a period meter. The period signal from this channel in usually quite noisy, since it is taken from the derivative of a counting rate, which usually shows cossiderable fluctuation.

Linear-Level Channel:

Sine neutron flux and reactor power are proportional, a convenient method of measuring power is to calibrate the current from a neutron sensitive ionization chamber. A gamma-compensated ion chamber is used in most reactors. The output current from a typical chamber of this knid in linear with neutron flux over atleast a 10^6 -fold rangethus, iF the current corresponding to a given power level is established, other powr levels can be determined by measuring the output current. the currrent FROM the I.ch. is usually measured by means of a galvanometer provided with shunts or a microammeter. The outeput is also sometimes displayed on a strip -chart recorder so that a permanent record of the power level is obtained.

Log n and period channel.

The power level or neutron flux varies over several decades in going from startup to full power. By means of the log n channel, a continuous indication of reactor power can be obtained. A block diagram of this instrument is given in Fig. 1. The voltage across the diode is proportional to the logarithm of the current from the F.ch, and

since current and neutron flux a density are proportional, a signal proportional to the logarithm of the power is obtained. this output is usually displayed on a log n meter and frequently on a strip-chart recorder. The period signal is used in conjunction with the softy circuits, too short a period, usually (5-10) sec,will resuet in a rapid shutdown (scram) of the reactor.

Control Room Instrumentations

Control Room Instrumentations are used for measuring and Control the process parameters. They are important for continueously monitoring the process parameters, so that they can be continoously controlled. They have no direct effect on control of reactor operation. Using these parameters, the thermal power can be easily checed.

Each process parameter measuring equipment is divided into two parts: The primary device (sensor and transducer) and the secondary device (indicator) that measures process parameters such as: temperature and temperature difference,' leak in pumps or pump room, water level, pressure, water flow and rarefaction. The primary devices are connected to the secondary devices through ordinary cables.

Power control system consists of seven loops as shown in Fig.1 three of them are for safety (S1, S2. and S3), one automatic control (C) and three for startup redundant loops (P_1, P_2 and P_3). Each of these loops has an ionization chamber (I.Ch.) ; either compensated or uncompensated.Figure illustrates the position elevation of I. Ch. inside the reactor,Figs.(2.a, 2.b and 2.c) illustrate the components of the three loop types. Each loop ends with signalling system and safety actuation systems.

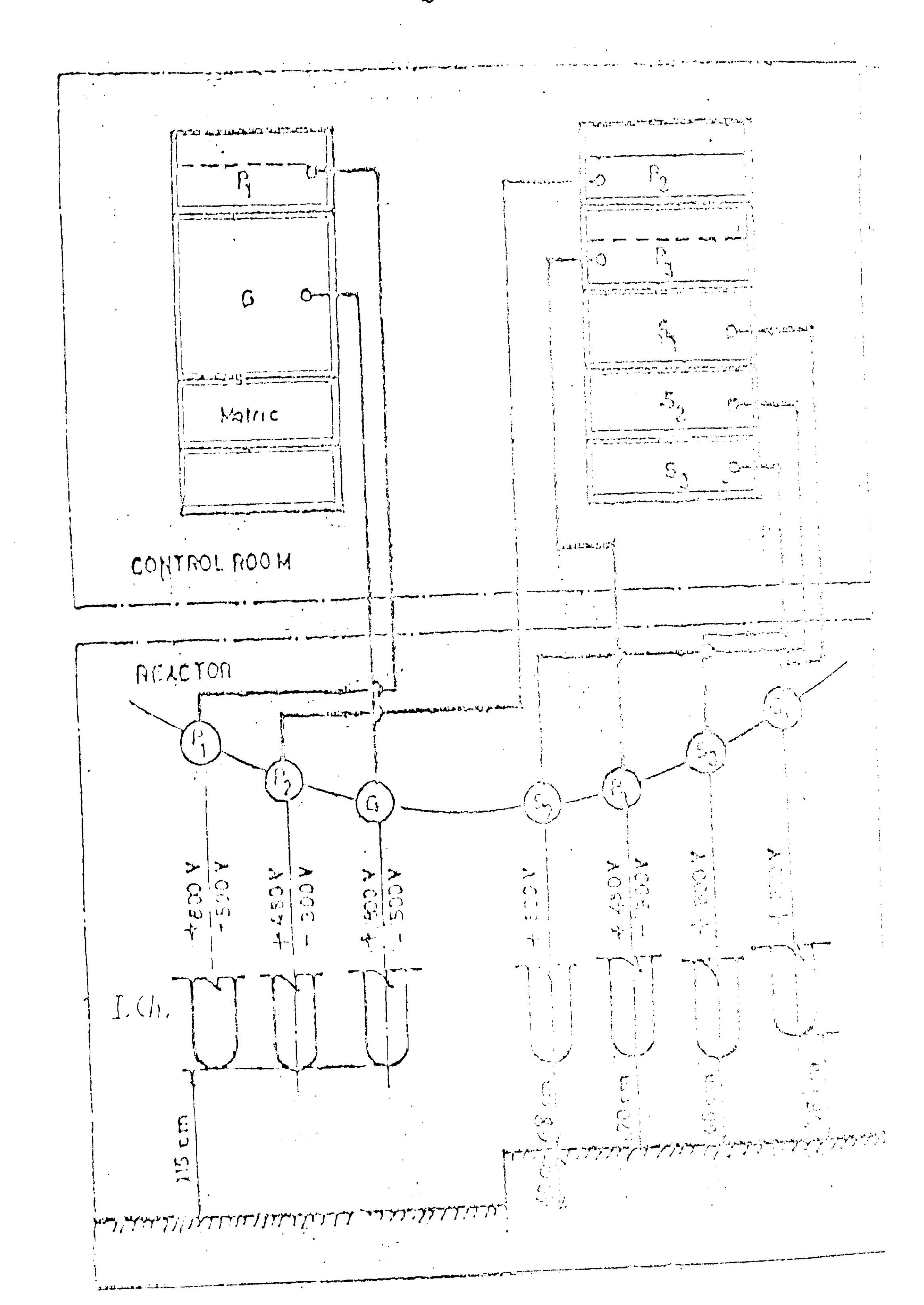


FIG (1) POSITION OF I.ch . INSIDE THE REACTOR

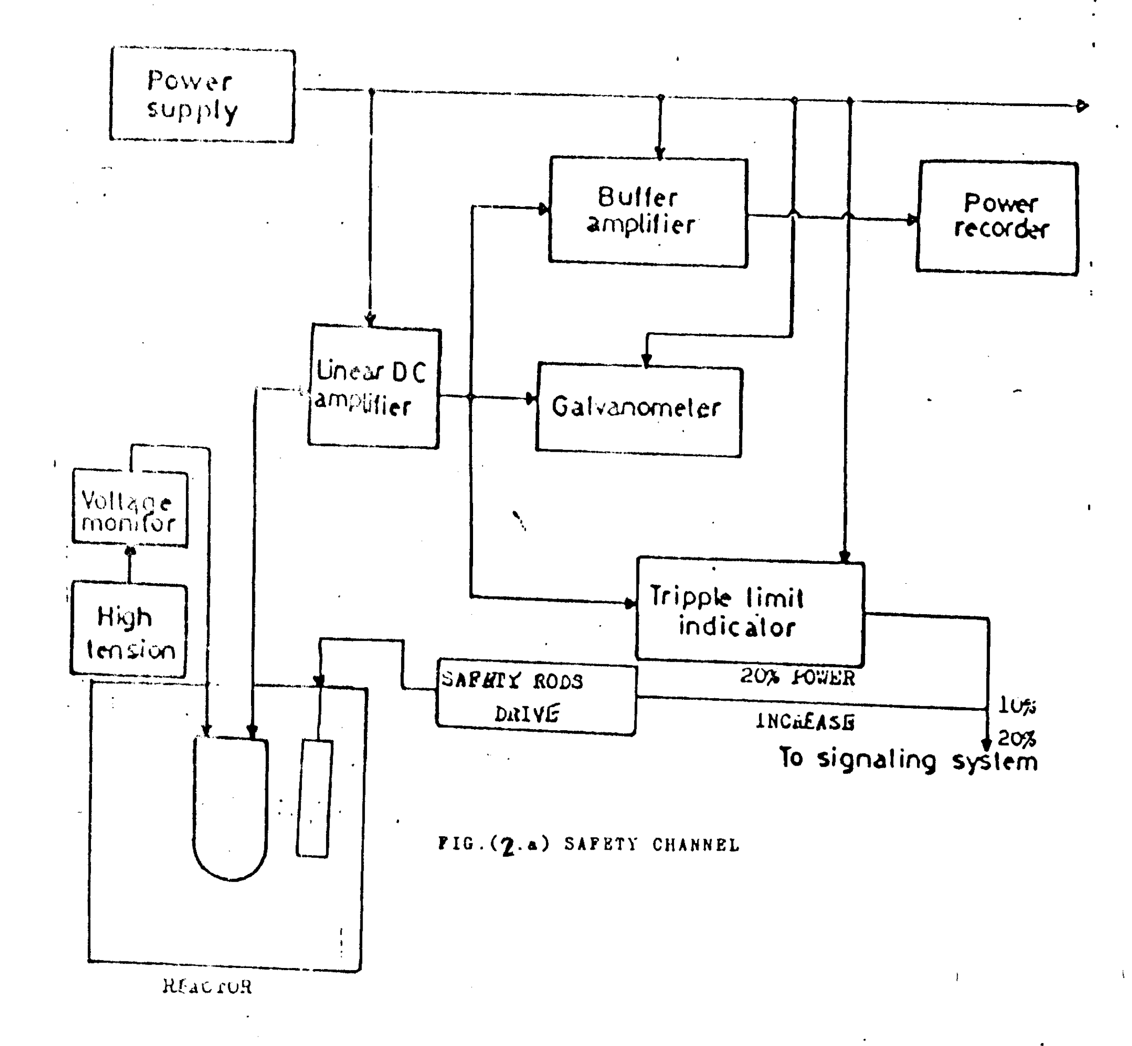


FIG.(2.a) SAFETY CHANNEL

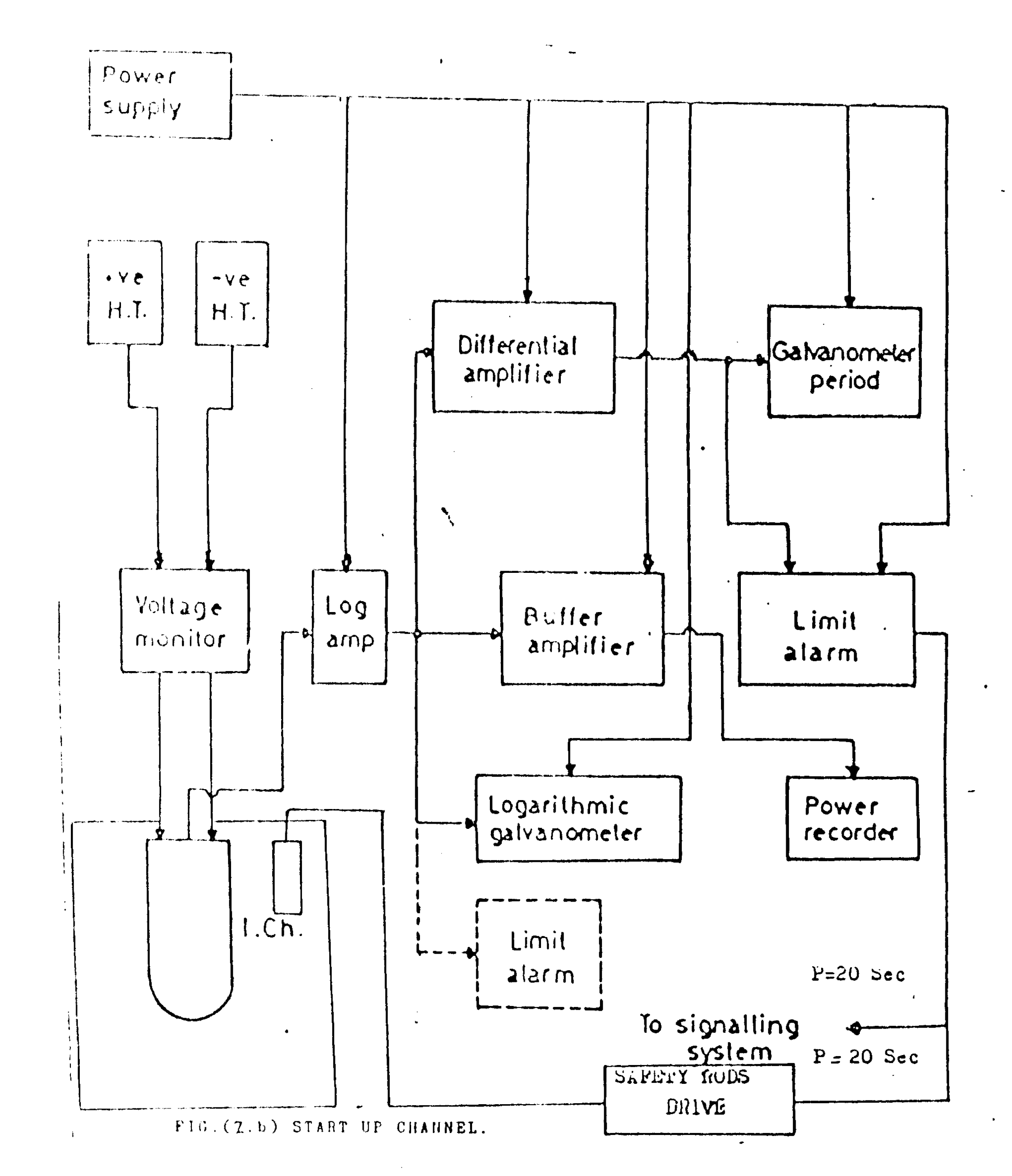


FIG.(7.b) START UP CHANNEL.

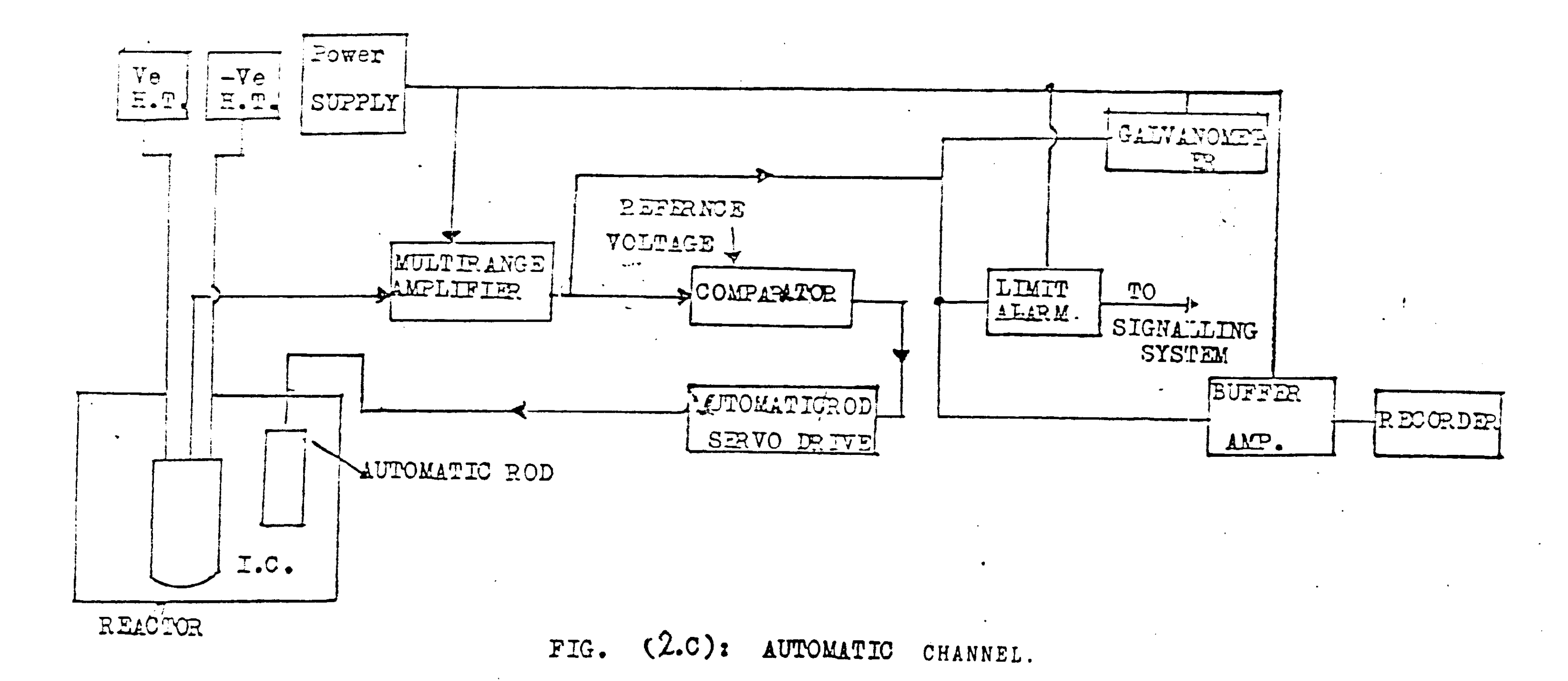


FIG. (2.C): AUTOMATIC CHANNEL.

PP2
PP1
AP
AZ2
PP2
AZ3
AZ1
PP1
Π P

First Shim Manual Rods PP1́ and PP1̋
Second Shim Manual Rods PP2́ and PP2̋
Safety Rods AZ1, AZ2 and AZ3
Automatic Regulating Rod AP
Precision Regulating Fine Rod Π P.

DISTRIBUTION OF THE CONTROL RODS IN THE ET-RR-1 REACTOR.

FISSION CHAMBER ASSEMBLY

NFA - 02.05

Features :

- Single fixed position fission chamber
- Neutron flux and period measurements
- 10 to 10 pps range
- Mean square range of 6 decades
- DC current range of two decades
- -3 s to +3 s period range
- Adjustable trip signals

The NFA-02.05 type Fission Chamber Assembly is a fixed position fission chamber probe with associated electronics. It provides neutron monitoring from start-up through the power range of a research reactor (Fig. 3).

The principle of the measurement

A single NFA-02.05 channel consist of a fixed fission chamber sensor, dry tube, qualified cable, preamplifier and electronics which utilize counting, mean square voltage and DC current technics. A graphic representation of the neutron flux monitoring function is shown below (Fig. 4.).

General description :

In the source range the neutron density is so low that individual neutrons are counted. A fission chamber is a U-235 coated ionization chamber designed to deliver a current pulse for each ionizing event. By using counters the count from the desired events must be separated from count of undesired events.
The pulse generated in the chambe must be amplified for transmission. Pulse height discrimination is one of the simpliest methods of separating wanted pulses from unwanted ones.
The mean square voltage method depends on the fact that if the time distribution of pulses from a nuclear radiation sensor is a Poisson distribution, the variance is a direct measure of the mean. This method has two advantages : increased gamma discrimination and more efficient use of neutron sensors.
The upper end of the neutron flux is measured by sensing the DC current of the fission chamber. In this range the detector works like an uncompensated ionization chamber.
Differentiating the log power signal we obtain the reactor period T, which is the reciprocal of the change in the neutron population per unit time.
The alarm unit provides signals at values such as high count rate, low period time or loss of chamber high voltage.

The signal of the fission chamber gets to the input of the preamplifier located in the shielding of the detector. The transformer coupling at both input and output allows for ground returns. The preamplifier output is connected to the input of the main amplifier.
The main amplifier has two signal paths. One for the pulse and the other for the MSV processing. An integral discriminator separates wanted and unwanted pulses. The MSV processor consists of a band pass filter and an 2 converter.
The DC current is measured in the positive path of the high voltage generator by using an isolated V/f converter.

Software controlled counters receive the output pulses of the discriminator and DC current measuring block. The A/D converter receives the output signal of MSV unit and the divided signal of the high voltage generator output. The high voltage power supply biases the neutron detector. Digital processing hardware controls the operation of the whole assembly, controls the counter, calculates the power and makes analog and digital output signals.
Operator interface is accomplished via a 2 x 20 character wide vacuum fluorescent display and a keyboard of 7 pushbuttons.
By means of a keylock the operator can determine the operating mode. (OPER/PROG)
In PROG mode one can set time constant of the ratemeter and the RMS conversion by means of the keyboard.
In OPER mode the equipment provides alarms (trips) when period and/or neutron flux level exceeds the set point value(s) and the operator can display all the measured and calculated values (pulse rate, MSV, DC current, power, etc.).
ALARM or WARNING lamps on the front panel lights if any of the alarm or warning situations has occured.
Watch-dog unit supervises the proper operation of the whole digital processing hardware. The WORK lamp shows the state of the watch-dog unit. The unit monitors the value of the high voltage power supply and the program execution.
The NFA-02.05 provides isolated analog and digital output signals for the control console and for the safety logic.

Construction

The instrument is constructed of standard plug-in modules housed in a chassis, designes for slide mounting in a standard 19" rack (Fig. 5.).The chassis front panel measures 19 inches wide and 4U (U=44.45 mm) high and the depth behind this panel is 440 mm. All electronic components are mounted on the plug-in modules which load from the top of the instrument. Maintance is accomplished by modul replacement when the instument is in its fully withdrawn position. Signal and power interfaces are on the back side of the rack.

The instrument consists of the following modules and main parts:
Mounting rack: mains power supply, keyboard, keylock, lights, preamplifier connector, PC/AT compatibile passive backplane.
Preamplifier (NFP-02.10) : Low noise, high speed amplifier.
Main amplifier (CMA-01): input amplifier, pulse discriminator,

msv amplifier bandpass filter, test generator, isolated output stage.

High and low voltage power supply unit (CPS-01): high voltage generator, isolated amplifier for measuring dc current, isolated low voltage power supply.

A/D & D/A unit (PCL-718): 16 ch 12 bit A/D, 2 ch 12 bit D/A, 16 ch DI, 16 ch DO.

Trip & range control card (TRC-02) : hardware period & power trip units, range and measuring mode control stage.

CPU: AT 286, 16MHz, with 80287-10 math co-processor possibilty, 2 RS 232 parallel printer port, battery-backup, real-time clock and calendar, 1MB DRAM

Counter (PCL-830) : 10 ch 16 bit counters, 16 ch DI, 16 ch DO

Relay output card (PCLD-785) : 16 relay outputs

RS 485 unit (PCL-743, optional) : 2 RS 485 interfaces.

Performance specifications :

Detector	:fission chamber type KNT-54
High voltage	500 V / 1 mA
Ratemeter -range	:10 to 10 pps
-accuracy	:2 %
Period meter	
-range	:-3s to 3 s
-accuracy	:2 %
Isolated outputs	
-analog (Power, Period)	
-voltage	:0 to 10 V
-load	:min. 1000 Ohms
-relay (Work, Power Supply OK, High voltage low, Power high, Per.time low)	
-rating	:100 V a.c/0.4 A max
-isolation	:500 V DC min

Front panel displays and signals

-Displayed values	:Pulse rate :MS value :DC current :Period time :High voltage :Discrimination level
-TRIP signals	:Low power :High power :Period time low :High voltage fault :Power supply fault
-Work signal	:High voltage available & program is running

Suggested operating conditions	:20 C to 30 C :220 V + 10 %
Temperature limits	:0 C to 50 C
Power requirements	:220 V 50 Hz , max 100 VA
Dimensions	width : 19" height: 4U depth : 440 mm

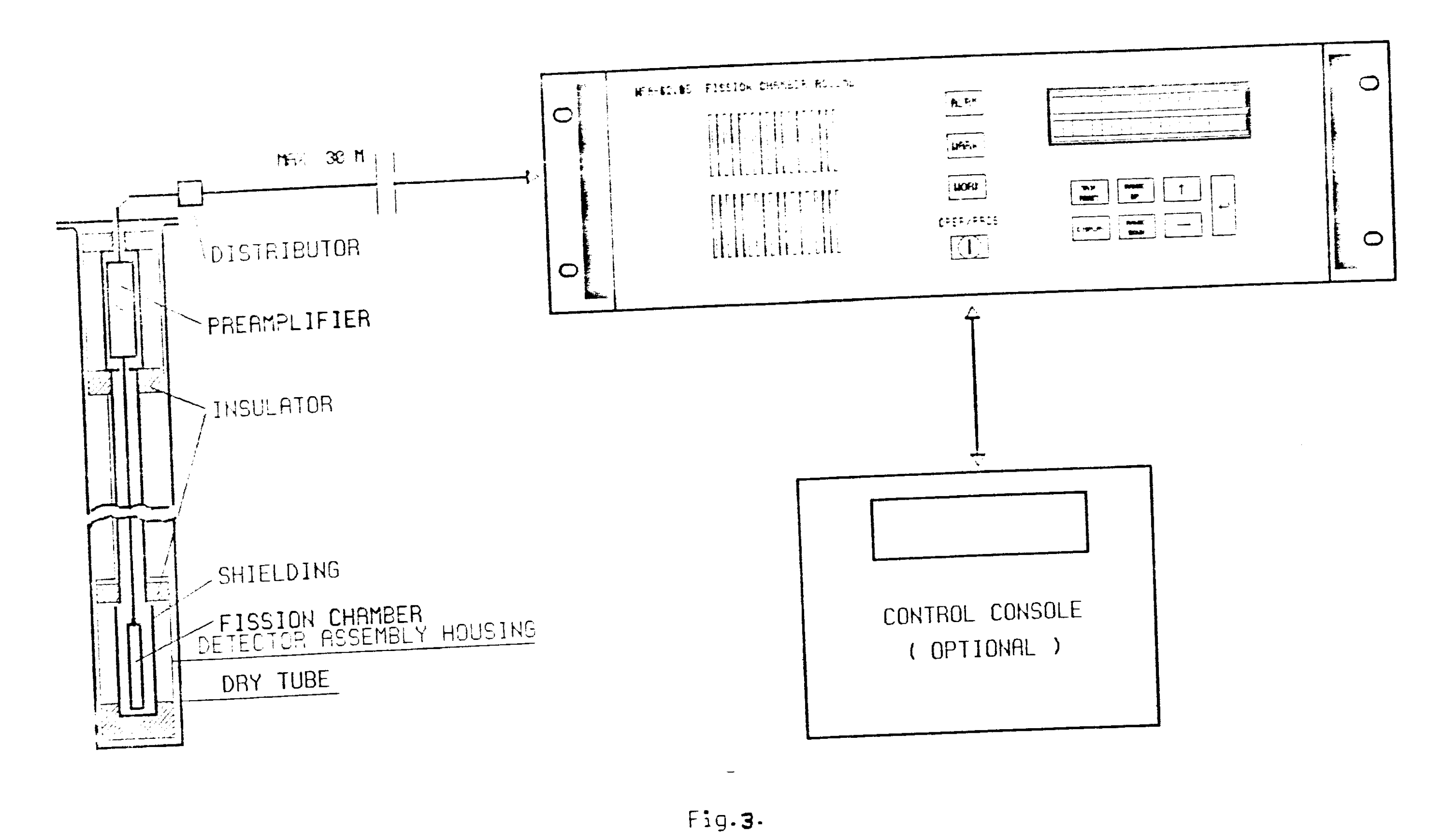

MAX 30 M
DISTRIBUTOR
PREAMPLIFIER
INSULATOR
SHIELDING
FISSION CHAMBER
DETECTOR ASSEMBLY HOUSING
DRY TUBE
CONTROL CONSOLE
(OPTIONAL)

Fig.3.

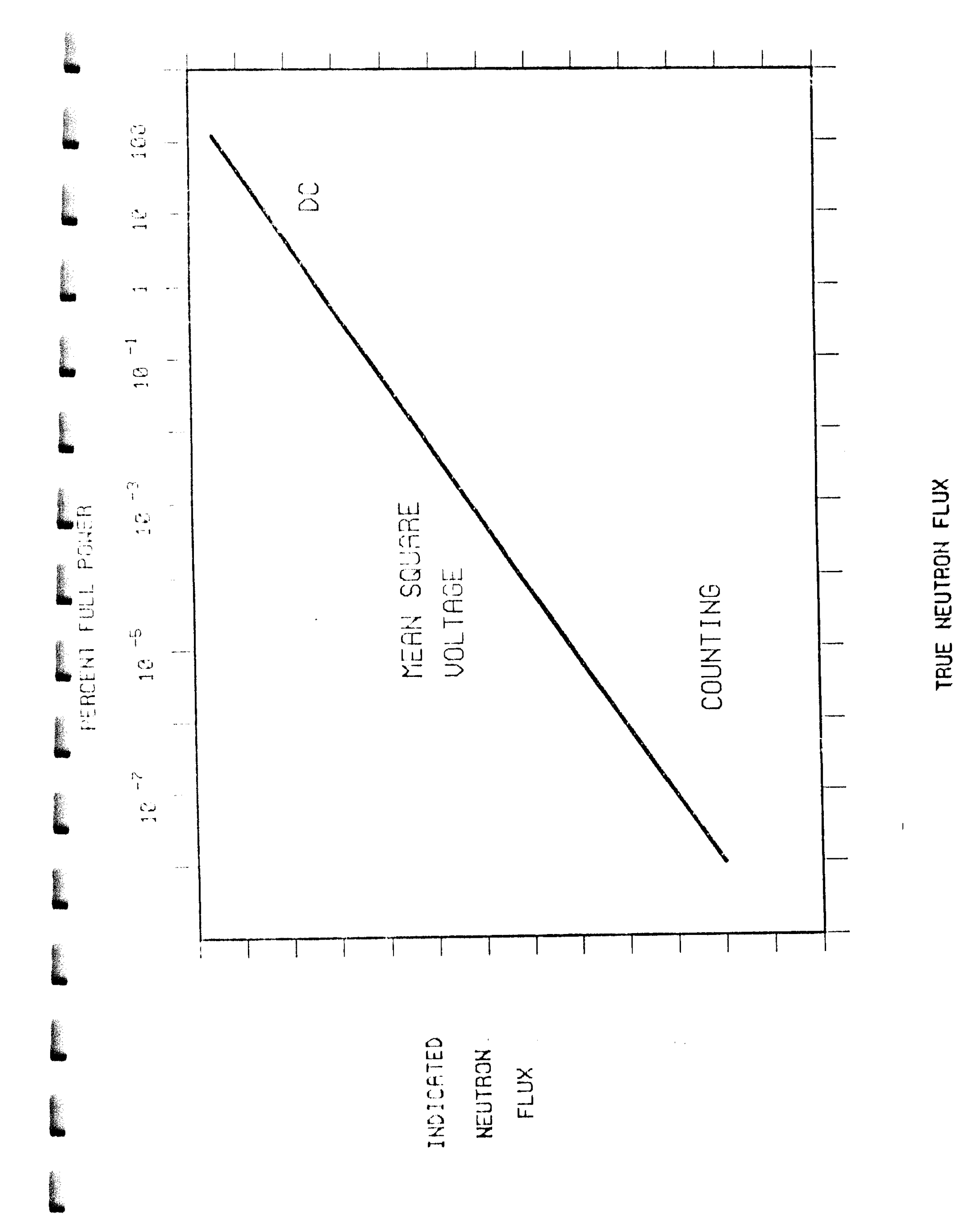

PERCENT FULL POWER
100
10
1
10 -1
10 -3
10 -5
10 -7
DC
MEAN SQUARE VOLTAGE
COUNTING
TRUE NEUTRON FLUX
INDICATED NEUTRON FLUX

FIG. 4

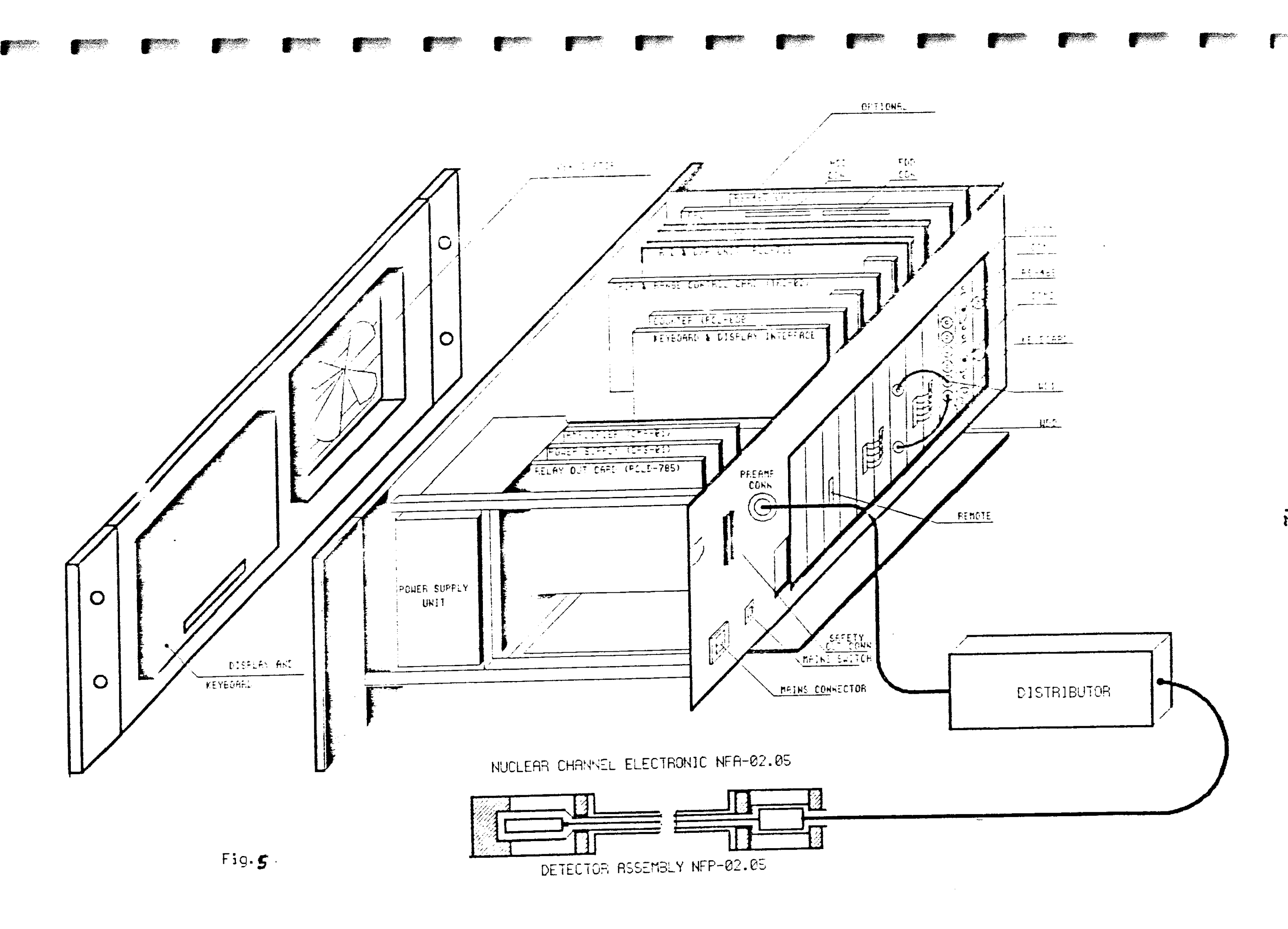

OPTIONAL
KEYBOARD & DISPLAY INTERFACE
RELAY OUT CARD (PCLE-785)
POWER SUPPLY UNIT
PREAMP CONN
REMOTE
SAFETY
MAINS SWITCH
MAINS CONNECTOR
DISPLAY AND
KEYBOARD
DISTRIBUTOR
NUCLEAR CHANNEL ELECTRONIC NFA-02.05
DETECTOR ASSEMBLY NFP-02.05

Fig. 5

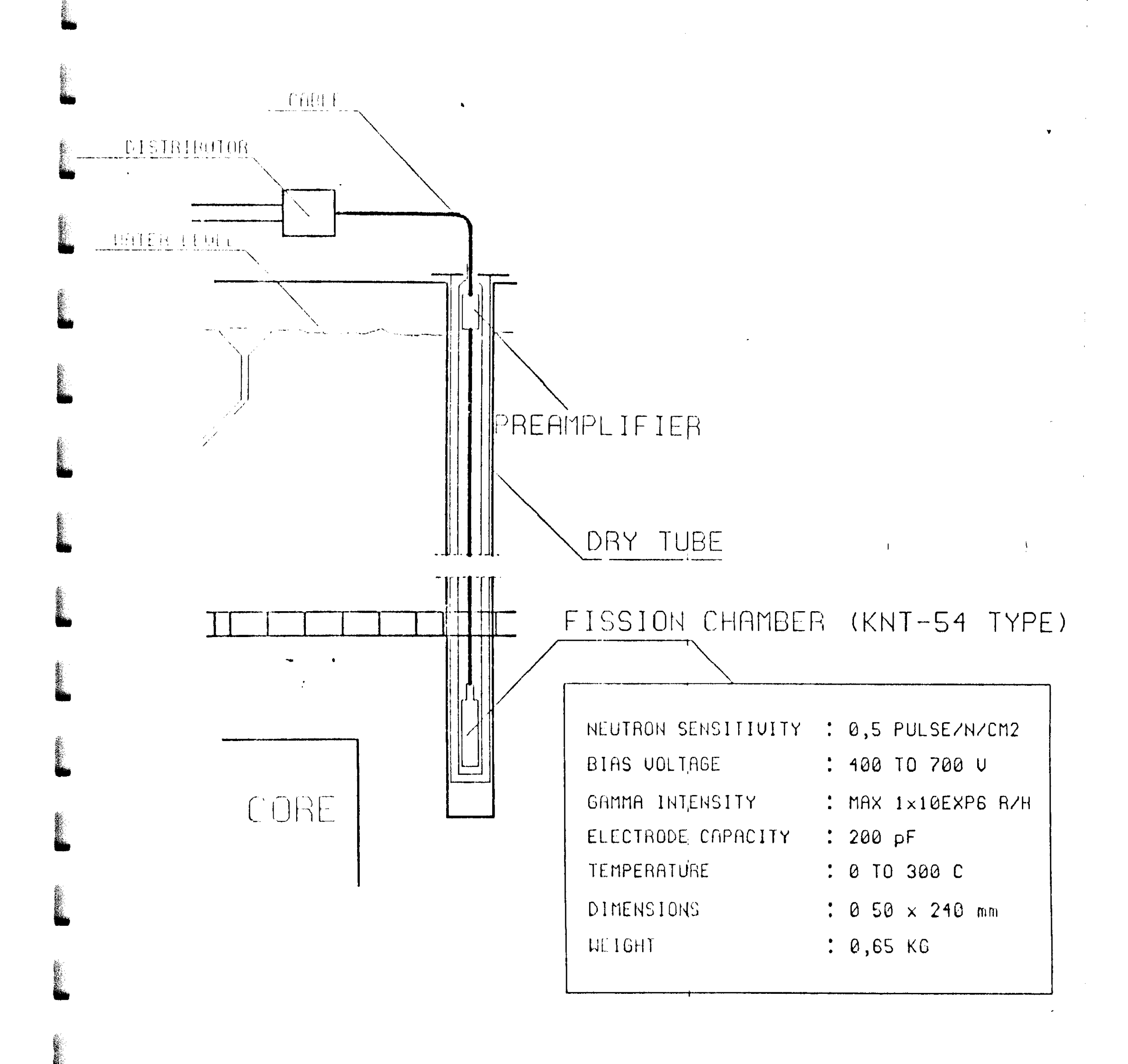


Fig. 6 FISSION CHAMBER PROBE

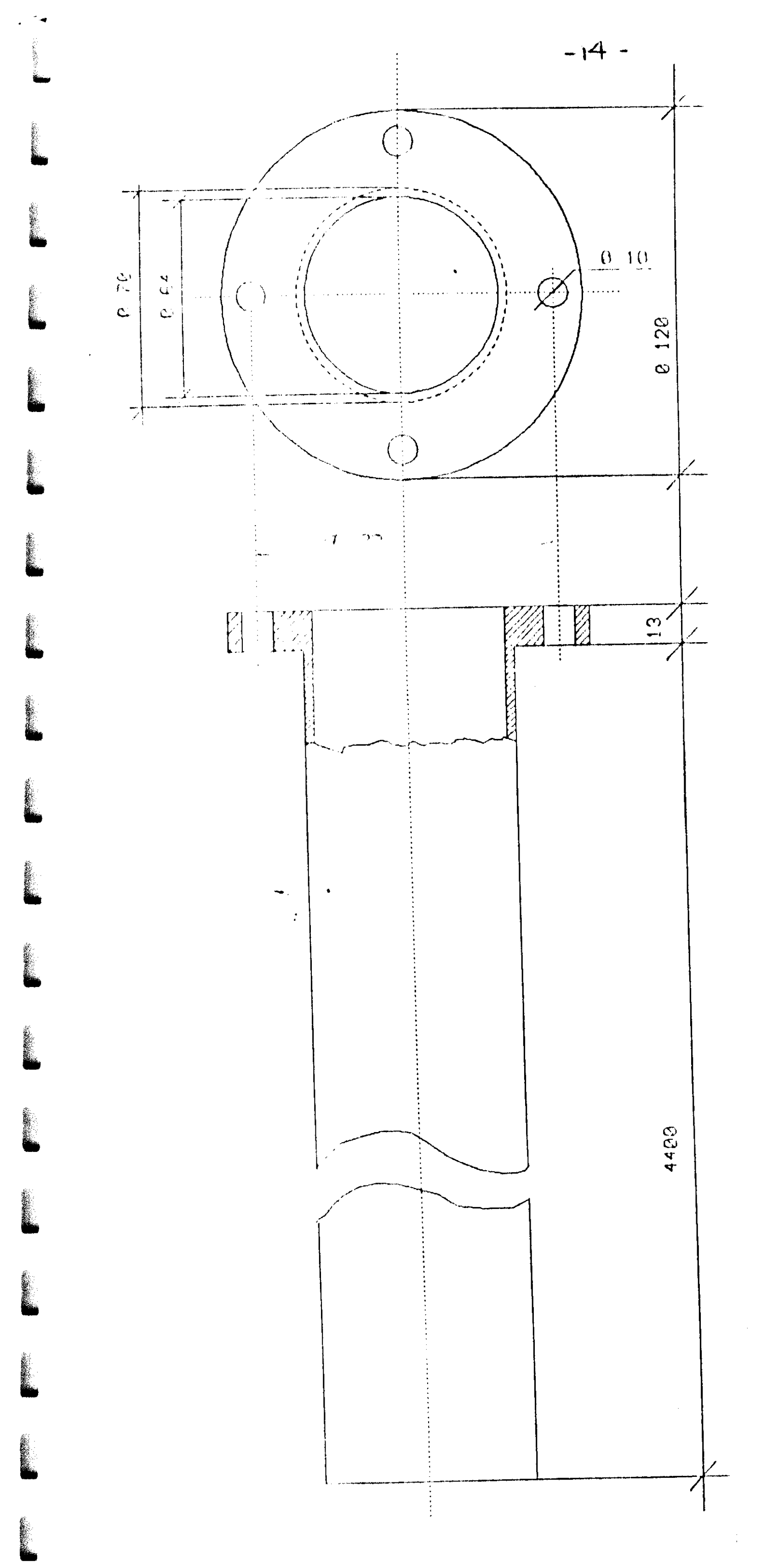


Fig. 7. DRY TUBE

PHILOSOPHY OF SYSTEMS RENEWAL

1- System electronices are based on integrated circuits and logic gates.

2- System units are of modular type. It can be used in different channels to minimize number of spare parts.

3- Standard output to computer for data logging, analysis and calculations.

4- System design fullfil safety requirements, standardization reliability, flexibility of servicing and repair.

5- Preparation of the electronic lab with new equipments for maintenance.

SYSTEM DESIGN



The parametric measuring system consists of 31 channels for measuring level, pressure, conductivity, flow, temperature power and safety indicators for reactor system.

<u>Pressure and Depression Channels</u> (Fig. 2.a)

a) Transmitters of type C719 are used to measure pressure or depression. The transmitter consists of two chambers separeted by a membrane. An output current 4-20 mA is produced linearly proportional to pressure difference across the membrane. In pressure channels the high pressure chamber is connected to the system while in depression channels the low pressure chamber is connected to the system. The other side is kept atomsphric.

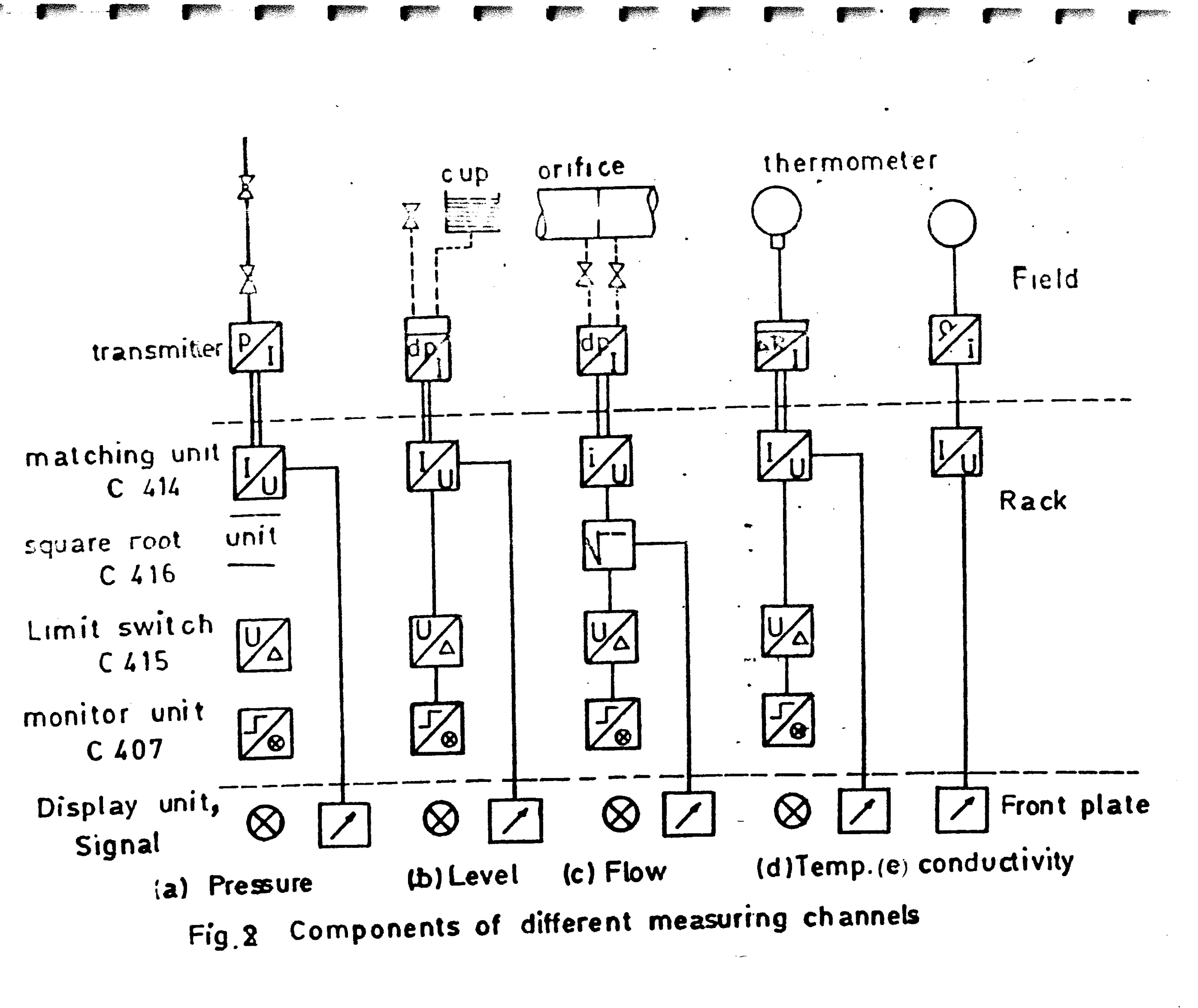


Fig.2 Components of different measuring channels

b) Transmitter output is fed to the electronic circuit C414 It is operated by 24 volt. Its function is:

1- Producing 30 volt for for operating the transmitter electrical circuit.

2- Producing voltage signal 0-10 volt corresponding to transmitter output current 4-20 mA.

3- Producing current signal 0-1 mA proportional to transmitter signal which is fed to a display unit that indicates the measured parameter.
One unit of C414 can be connected to two channels.

c) Voltage signal output of C414 is fed to the limit switch unit C415 which acts as a comparator. A limit value is adjusted by a potentiometer and compared by the input signal. The difference operates a reed relay. The output of the unit is a two state free contact. C415 is used for minimum or maximum adjustment. The unit has 4 identical components and can be connected to 4 channels.

d) Monitor unit C407 use the potential free contacts of C415 to do the following functions:

1- initiating signal to a sound generator to give an accoustic warning

2- initiating signal to be fed to a lamp indicating an abnormal situation.

3- A memory circuit will produce a blanking signal for the first coming one so that the operator can determine which signal cause the abnormal situation.

Level Measuring Channels (Fig. 8.b)

a) Water level in a tank is measured as a pressure difference between tank water column and water cup installed at the base of the tank. Transmitter type C719 is used to measure pressure difference proportional to water level. At maximum level an output of 20 mA is produced.

b) The electronic part is similar to the pressure channel.

Flow Measuring Channel (Fig. 8.c)

a) The flow rate is measured by determining the pressure drop across an orifice meter where:

$$Q = C \left[\Delta P \right]^{0.5}$$

The constant C is function in orifice design. Transmitter type C719 is used to measure the pressure drop.

b) The electronic part is similar to the pressure channel in addition to a square root extractor unit C416 whose input is 0-10 volt and a matching unit C414 whose output is 0-10 volt. The output Y is related to the input X by the relation:

$$Y = A \left[X \cdot 10 \right]^{0.5}$$

where A is an adjustable parameter 0-1.0. Thus Y is proportional to the flow rate.

Temperature measuring channel (Fig. 8.d)

a) The temperature is measured by platinum resistance thermometer through Heutston Bridge that produce voltage to a millivolt converter containing an amplifier. Current signal 4-20 mA is produced linearly proportional to temperature change.

In view of temperature difference two platinum thermometers are embedded in hot and cold legs.. The output of the amplifier is proportional to the difference in temperature of both legs.

b) The electronic part in control room is similar to the pressure channel. It contains matching unit C414, limit switch C415, monitor unit C407 and a display unit.

Conductivity measuring channel Fig. 2.e

a) Water conductivity indicates its impurity. A probe of type Rosemount 142 is immersed into reactor coolant circuit to measure its resistance. Transmitter type 2101C RAMOVILL converts the resistance change to an output current 4-20 mA.
b) The electronic part contains a matching unit C414 and display unit of range o-20 uS/cm without alarm.

SIGNALLING SYSTEM

The output of the monitor unit C407 of each channel is connected to light and accoustic signalling system. It has the following characteristics:

a) Light signal is generated in case of an abnormal measured value. It remains unless its default is recovered.
b) An accoustic signal is generated for each light signal.

c) Three signals are directly connected to the emergency circuit to scram the reactor. They are:

1- 20% pressure reduction in primary coolant circuit.
2- 10% flow reduction in primary coolant circuit.
3- 20% reactor power increase.

AUTOMATIC CHARGING SYSTEM

The principle of guarding against unwanted events by providing successive protective barriers is frequently called defense in depth. Defense in depth has been a useful concept for addressing safety in current light-water reactors. The components that present defense in depth for ET-RR-1 in particular to reactor coolant system are:

1- Water level signal indicator at an arbitrary setting value of 500 cm in the reactor central tank.

2- Feed water supply system of capacity 40 cumeter equal to reactor and coolant circuit capacity, Fig.3. The water is supplied from a deminralizer plant to the feed water supply tanks (FWST).

Therefore, it is decided to increase that degree of defense in depth by installing an automatic charging for the (FWSS). It is mounted in the system in parallel to the main feed water valve, Fig. 9. The advantage of that system is to minimize considerably the time consumed for replenshing during level decrease between the upper setting value, 580 cm, and the alarm level at 500 cm above the core, where reactor safety limits are specified. Its signalling system indicates the operational mode (manually or automatic) and the valve status (open or close).

Philosophy of replensher design

1- The system has the flexibility to be operated automatically from control room or manually from reactor hall. Therefore, the time consumed to fill the shield tank in advance is eleminated. Also, the problem of loss of electrical power supply or motor and limit switch malfunction is bypassed.

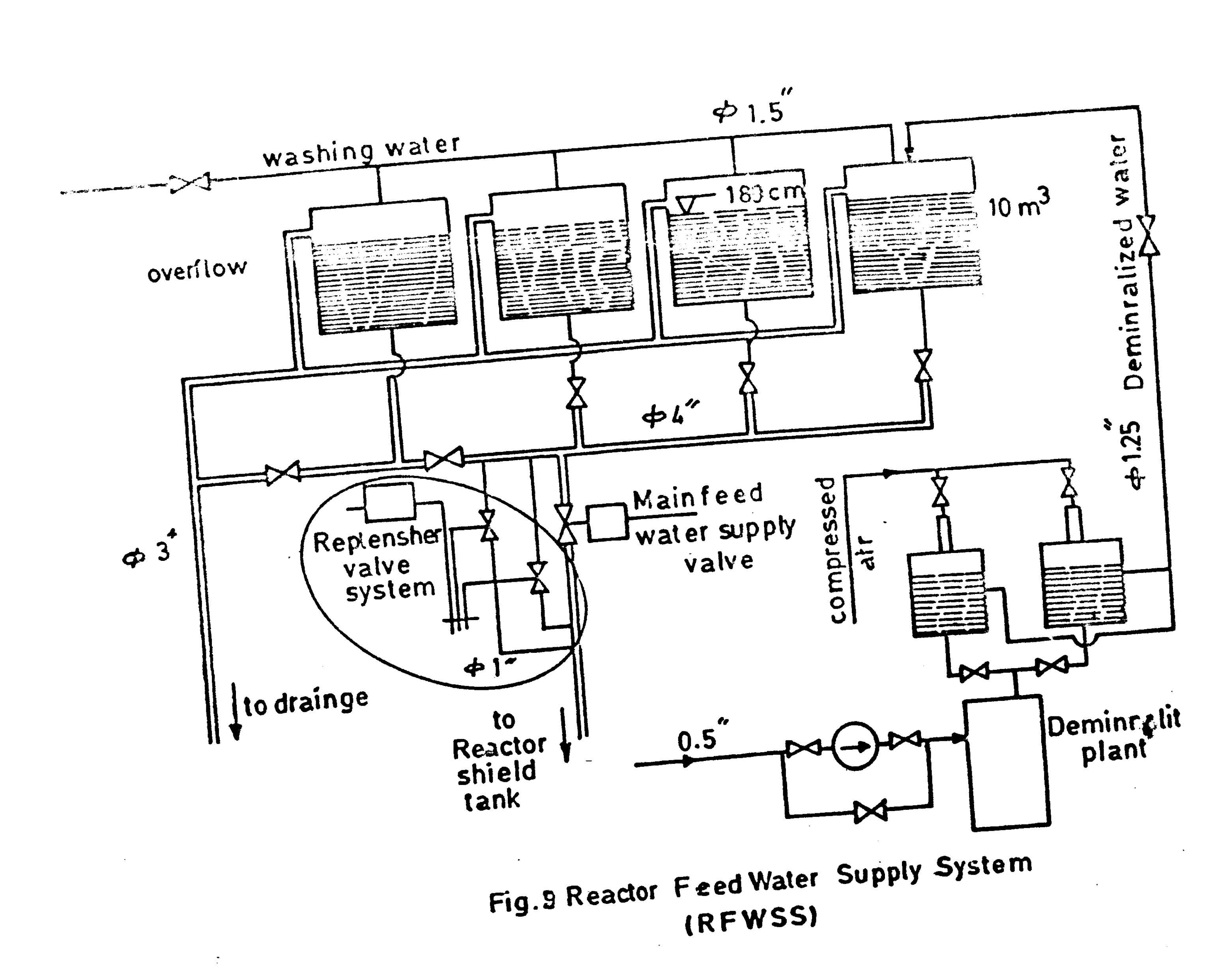


Fig.9 Reactor Feed Water Supply System (RFWSS)

2- Appropriate decision with regard to the main feed water supply valve is taken in time according to water level status in reactor central tank particularly in case of loss of coolant accident.

3- Saving of over flow water loss.

Construction and operation

In the reactor hall two stainless steel pipes of diameter 1.25 inch are welded between the common feed water pipe and the pipe leading to reactor body in parallel to the main feed valve of 4 inch diameter. The valves are driven by an actuator. It consists of motor (4 watt, 1300 rpm) and two geare for speed reduction. The driver gear is mounted on ball bearing while the other is mounted on bronze socket. A break is mounted on the shaft to stop the motor at the end of the valve stroke. Fig. 10. shows the supply circuit of the motor is fed with 220 volt A.C. from the same supply that provides the central tank level measuring channel. Two contacts KPO and KP2 rotate the motor in the open and close directions respectively. End switch SO at low speed shaft is adjusted to stop the motor and signalize the valve open state. End swith SZ is adjusted to stop the motor and signalize the valve to cloce state.

Inside the control room a control unit is installed in the G356 panel which include the central tank level measuring channel. It consists of the following components:

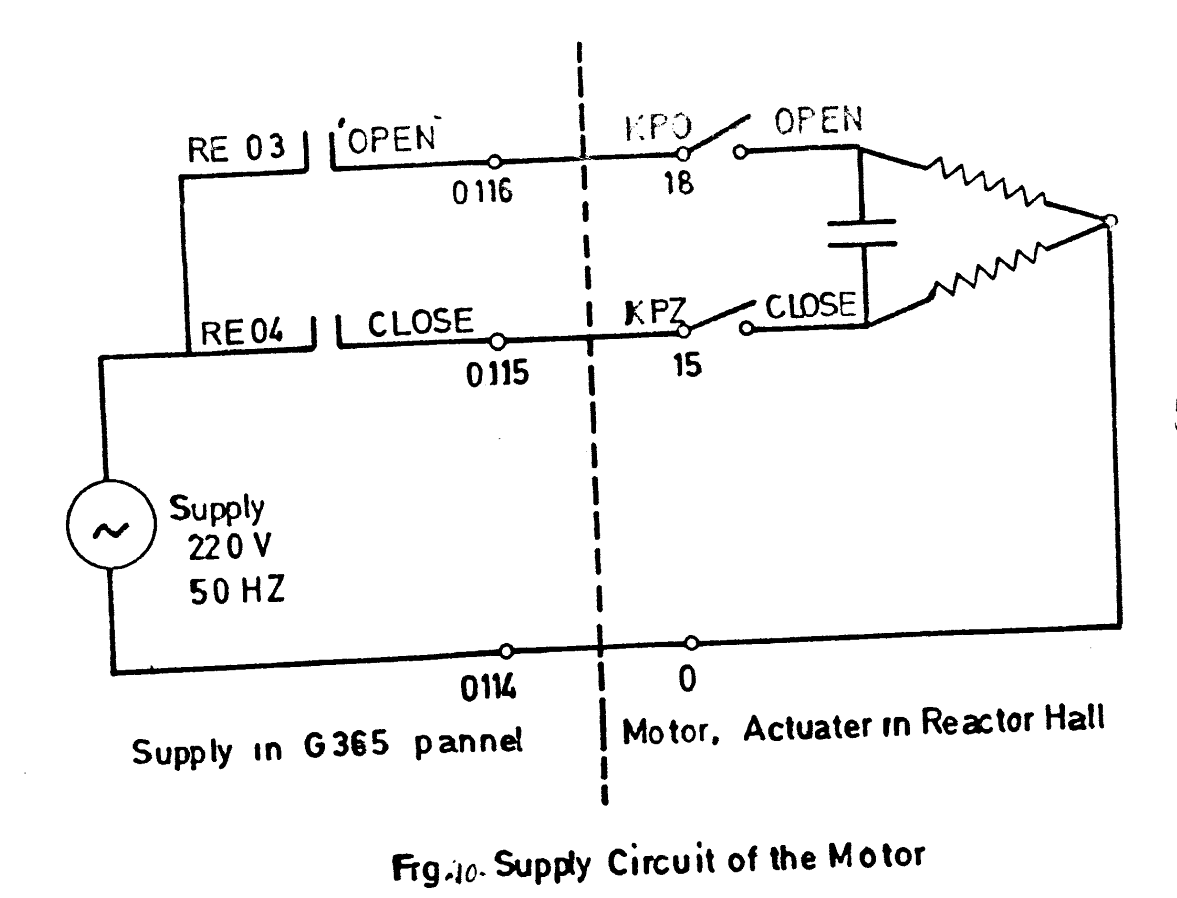


Fig.10. Supply Circuit of the Motor

1- Push button Man/Aut. To choose manual mode of operation the button is pushed and the lamp indicates manual operation. In case of automatic mode the button is settled and the lamp is off.

2- In case of manual operation, two push buttons OPEN and CLOSE are supplied by 220 volt A.C. to the motor in the desired direction. At valve end positions the lamps signalize its state.

3- The automatic operation of the system is controlled by the limit switch unit C415 of the central tank level channel L002. If the water level decreases below 560 cm the switch will connect the supply to the motor in the open direction to feed water to the reactor. If the level exceeds 580 cm the swith connect the motor to the close direction. The setting points and the hysteresis can be changed from the C415 unit.

4- Relay RE03 on the open direction line is energized if the the unit opens the valve. It has a contact on the 220 volt side of the motor.

5- Relay RE04 on the close direction is energized if the unit closes the valve. It has a contact on the 220 volt side of the motor. RE03 and RE04 can't be energized at the same time so that the motor can be supplied only from one side. This is done by closing contact in each line from the other relay. The control unit operates by 24 volt DC from the DC supply unit of G356 panel. Fig. 11.

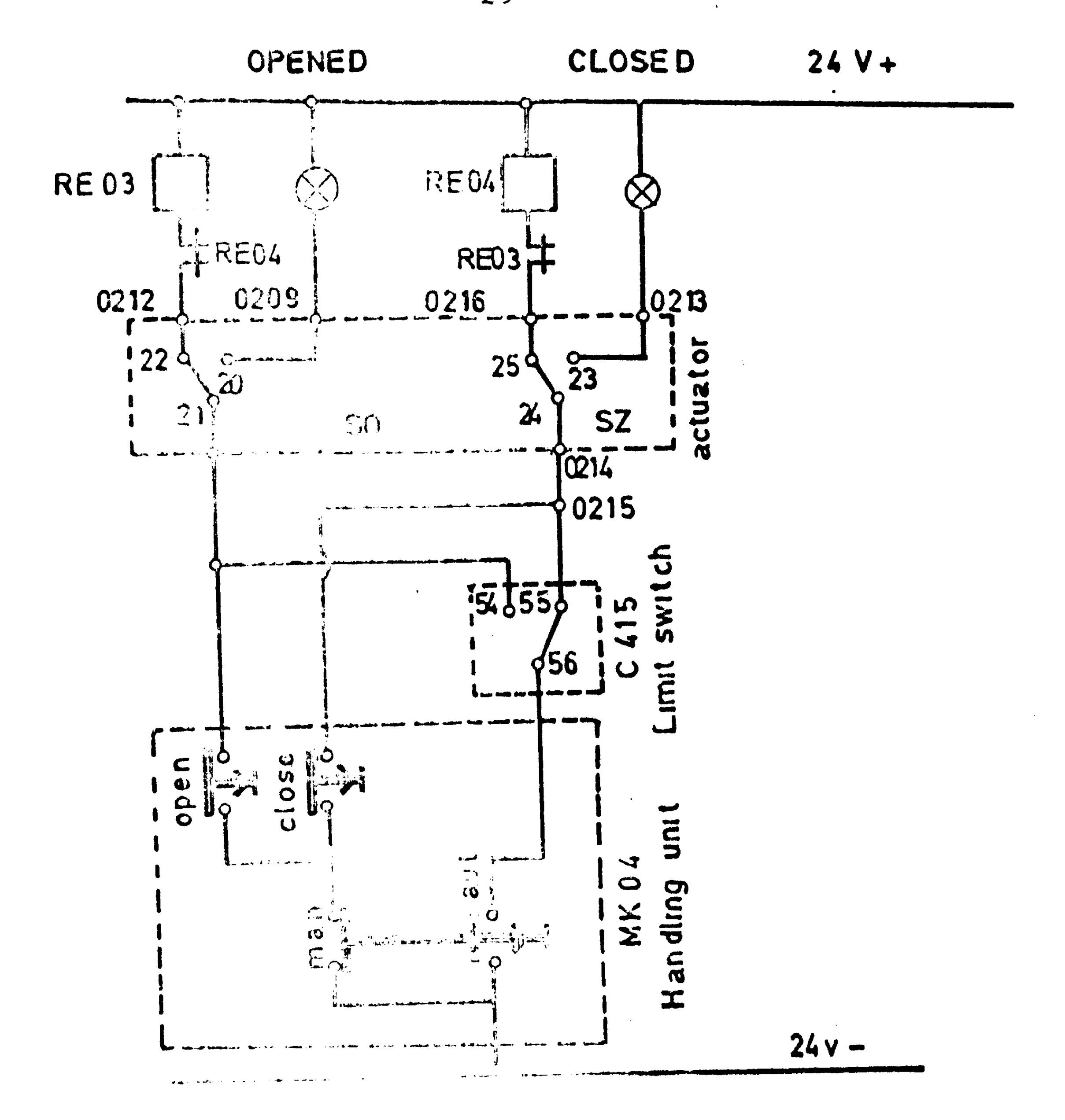


Fig. 11 Design features of the control unit

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REACTORS DEPARTMENT

MODERNIZATION OF ET-RR-1 MEASURING SYSTEM

BY
F.H. DIMITRI AND M. KHATTAB

1991
INFORMATION AND DUCUMENTATION CENTER
ATOMIC ENERGY POST OFFICE
CAIRO, A.R.E.

Lag.5.2

REACTOR INSRUMENTATION (ET-RR-1)

BY

M.K.SHAAT

1. Nuclear Instrumentation and Control (I & C) System

1.1 General Description

The new I & C system is composed of the following sub-systems:-

- Safety channels.
- Logarithmic Channels.
- Linear Switchable Channel.

1.1.1 Safety Channels

The safety channels consist of three independent channels which monitor the reactor power. These are connected to the emergency circuit through a 2 out of 3 logic. Each safety channel has three (3) ranges (10^{-6} amp., 10^{-5} amp. & 10^{-4} amp) which ensure safe operation of the reactor at different power levels.

Fig. 1. presents a schematic of the safety channels showing its modular electronic components.

The old safety system was composed of only one channel in the start-up range and two chanels in the power range with one common powersupply. There was no buffer amplifiers or recorders on these channels.

1.1.2 Logarithmic channels

Three independent channels are used to measure the reactor period during the start-up of the reactor. These are connected to the emergency circuit through 2 out of 3 logic.

Fig. 2 presents a schematic of the logarithmic channels showing its modular electronic components.

In the old system there were no any logarithmic channels instead the doubling time was measured by the operator using a stop watch.

1.1.3 Linear Switchable Channel

This channel measures the reactor flux from 10^{-12} amp to 10^{-4} amp. in 16 ranges which allows the precise change in reactor power to the detected. This channel is connected to the automatic rod drive system through a control unit (Motric 96 E) to control the reactor power in the automatic mode of operation.

Fig. 3 presents a schematic of the linear switchable channel showing its modular electronic components. The Motric 96-E is an on/off switch operated from the input signal from the linear amplifier. It connects the voltage of the automatic control rod to the motor drive if the difference between the measured reactor power and the reference value exceeds $\pm$ 1%.

In the old system an amplidyne was used to amplify the difference voltage which operate the motor drive of the automatic control rod. A comparator using d-c supply from a dry cell was used to produce the reference voltage. The resulting transient response was not as good as in the new system.

From the above description of the rew I & C system it is clear that it included the following improvements compared to the old system:-

A- Independency: Through the use of independent ionization chamber, power supply and high voltage supply for each channel.

B- Reliability: Through the use of 3 channels for each system with a 2 out of 3 logic unit which enables on line testing.

C- Availability: Through the use of standard modular units and identical components in all units. This enables the operator to interchange the components in the three units without the need of too many spare components. Modular design and identical components made the maintenance a lot easier.

The safety of the reactor is greatly enhanced by introducing the above improvements.

1.2 Procurement and Installation

Through a bilateral agreement with KFA Julich, in F.R.G., the design of the new system was jointly developed through 1980. Hartman & Brown company in F.R.G. was selected as the supplier of the new system. Installation was carried out by ETRR-1 operation and control staff with the help of some experts from KFA Julich. Installation was completed in 1984.

2. Radiation Protection System (RPS)

2.1 General Description

The RPS is designed to cover the monitoring of radiation levels at selected areas in the reactor building. The system is composed of 30 channel gamma radiation assembly, together with an air monitoring and water activity controlling equipment. The distribution of these detectors in the reactor building is shown in Fig. 4. Twenty five channels of the RPS utilize silicium semi conductor detectors covering the different ranges of radiation exposure from 0 to 3×10^4 uSV/h in five steps. The other five channels utilizes Geiger-Muller detectors as gamma indicators for measuring the activity concentration in the sampled air. Each channel has an alarm output with adjustable level. Monitoried radiation levels are continuously compared with alarm set points for each detector channel. As the radiation level approaches the set points, alarm is initiated at the central unit. At the same time alarm signal appears at the detector itself which is equipped with alarm unit. The central unit of the RPS includes a series of electronic channels mounted on 10 racks. These racks contain alarm indicators, linear rate meters of the 30 channels, printer, switches of air monitors ect... Eight of the 30 signals are relayed to the operator in the reactor control room which is located one floor above the location of the central unit.

2.2 Procurment and Installation

The system was supplied by Hungary through IAEA technical assistance (project No EGY/9/015). Installation was carried out by the operation reactor staff together with the Hungarian experts. First the old system was dismantelled and the wiring completely changed to install the new system which was first operated in 1988. After operation of one year, recalibration and adjustment were under taken in march 1989.

The new system has a lot of advantages over the old system which can be summerized in the following:

- Standardization: the new system contains standard units so it can be replaced by any other similar units.
- Independency: each channel has its own detector, amplifier, rate meter and alarm system.
- Availability: the channel use identical units which enable the change of one unit by another unit from other channels.

The new system has greatly enhanced the radiological safety measure in the reactor building.

3. Process Instrumentation System (PIS)

3.1 General Description

The PIS consists of 30 channels to measure the process parameters. The sensors in the different locations transfers the parameters to current which is amplified and send a volt signal to 3 centeral units in the control room, where it is displayed and recorded. The parameters measured are:

- Primary coolant circuit: pressure, flow, temp., temp. difference, flow in filter, pump bearing temp.
- Dearator: water level, flow, air flow, outlet temp. depression.
- Secondary coolant circuit pressure, flow, temp., temp. difference..
- Levels: central tank, shield tank, distillate water tank, sprent feal storage, wast storage tanks 1 & 2.
- Air depression: above reactor, under reactor, in spent fuel tank in hot cells, pumproom, before ventilators.
- Thermal power indicator.
- Water conductivity in distillate water tanks.

The 3 central units contain the amplifiers, displayers and alarm signals for each channel as shown in Figures 5,6,7.

3.2 Procurement and Installation

The PIS system was supplied from Hungary through IAEA technical assistance (project o. EGY/4/028), in 1989. It is anticipated to be installed by the Hungarian experts and ETRR-1 operation and control staff during the last quarter of 1989.

The new system is a modular system having the same advantages mentioned before for the other two systems namely standardization, availability, reliability and independency.

4. Protection System

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

4.1 Protection System Functions

The protection system shall be designed 1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and 2) to sense accident conditions and to initiate the operation of systems and components important to safety.

4.2 Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

4.3 Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

4.4 Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that 1) no single failure results in loss of the protection function and 2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of the functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

System Description

The function of the automatic protection system, also named scram system, as producing (automatically or even manually) a fast decrease of power in the reactor when the integrity of the physical barriers containing the fission products and the fuel elements are threatened. The scram system consists of the following elements :

1- Sensors that monitor permanently the development of the physical parameters representative of the functioning of the reactor, either in normal or abnormal situations.
2- Equipments transforming the signal sent by sensors into measurable electric currents (amplifiers,etc...).
3- Comparators actuating a signal indicating that the physical parameters measured have exceeded the set value (threshold value).
4- Computerized safety logic system, which contains logic circuits grouping and processing the signal from the comparators and giving the scram signal.
5- Equipments called circuit breakers that from the scram signal directly actuate the servo-drive loop (emergency circuit) to insert the control rods into the reactor core by gravity.
6- Control rods whose function is the stop of the fission reaction in the core by fast absorption of thermal neutrons.

Actuation of the Protection System

The scram system is designed in order to respond to a long list of initiating events, which are considered as potentially leading to abnormal consequences or to accidental conditions. Each initiating event characterized by some measurable parameters that if exceed certain safety limits it will actuate a shutdown signal to the scram system which induce the control rods to fall down into the reactor core.

Flow of Scram Signal

Figure (14) shows the flow of each scram signal from left to right. Each signal measured by a sensor. The signal will transfer to a measurable electric current by the amplifier. This signal will be compared by a reference safe limit value. If this signal exceeds the safe limit value, the signal will be proceed into the CSLS and the scram order will be given to the control rod drives of the safety rods.

Success Diagram

Figure (15) is a schematic success diagram, from left to right, shows the sequence resulting from the scram signal followed by an effective scram. There are 27 signals, one of them must give scram order to the CSLS that in turn will disconnect the current from the control rod drive circuit and the safety rods will be inserted inside the reactor by gravity force and spring force.

Assumptions

1- The manual emergency shut down was not included in the fault tree, since it intervenes directly on the logical system.
2- The failure of 220V AC power supply was not included in the analysis, because it acts as a fail-safe-signal that will scram the system directly.
3- Because the CSLS is a fail-safe-system, as a result of the safety feature of the system,

Control Rod Drive System

The distribution of the nine control rods in the core is shown in Fig. 8. The main feature of the control rods including material as well as normal or insertion speeds are given in Table 2.

Table 2. Main Features of Control Rods

Control rod ident.	Purpose	Number	Material	Normal speed cm/sec	Emerge. speed cm/sec
AZ	Safety	3	B_4C	2	250
AP	Automatic	1	SS	0.5	3.5
nP	Fine regula.	1	B_4C	0.2	3.5
PP-1	Manual	2	B_4C	0.6	3.5
PP-2	Manual	2	B_4C	0.6	3.5

The control rod drive system has three types of circuits:

- Safety rod drive circuit
- Manual rod drive circuit, and
- Automatic rod drive circuit.

Safety Rod Drive Circuit

The three safety rods,(AZ1,AZ2, and AZ3) each has its own motor and associated circuit that does the following functions:

- Lifting the safety rod out of the core before reactor operation.
- Holding the safety rod at its upper position during operation.
- Pushing the safety rod down inside the reactor core to shut-down the reactor.
  The safety rod drive circuit has been designed such that it should not operate the motor to lift the safety rod unless the following conditions are fulfilled :
- The emergency circuit is ready for operation.
- The two other safety rods are not lifted simultaneously.
- The magnets that connect the 48 volt DC to the circuit and the motor are available.
- The safety rod must not be in its upper position.
- The motors driving safety rods are uni-directional (lifting only) with a speed of 2 cm/sec.
  In shut-down (either normal or emergency shut-down) the safety rods are dropped into the core by gravity within less than 0.2 sec.

Manual Rod Drive Circuits

The rod drive circuits of the manual rods consists of two pairs of rods PP 1 & PP-2. These rods are called also start-up rods. There is one drive circuit for each pair. The rod drive circuit of the manual rods is different from that of safety rods in the following points :

- The position indicator for the safety rods indicates only upper and lower positions, where that for manual rods indicates the actual vertical position in the core.
- The drive motor is reversible and multi-speed of insertion is much higher than that for lifting the rods.
- The manual rods do not fall under its weight even in an emergency shut-down.

The normal speed of the manual rods is 0.6 cm/sec.,and the insertion speed is 3.5 cm/sec.

Automatic Control Rod Drive Circuit

The automatic control rod (AP rod) drive circuit is very similar to that of the manual rods drive. It has an additional circuit that moves the rod up and down during reactor operation under the automatic control.

Reactivity Control

It is necessary for nuclear reactor to have a predetermined excess reactivity in the form of additional amount of nuclear fuel above the critical case to compensate for the following effects :

- Fuel burn-up
- System poisoning by Xenon-135, Samarium-149, and other fission products

The total amount of excess reactivity is equal to 0.05 $\Delta k/k$ is the sum of the following terms :

- Compensation of system poisoning by Xenon-135 = 0.025
- Compensation of fuel burn-up = 0.024
- Compensation of reactivity changes due to decreases in coolant temperature = 0.001

Control of excess reactivity is compensated by the control rod system. The measured reactivity worth of the control rods are[7] :

Rod type	AZ	PP-1	PP-2	AP	nP
($\Delta k/k$) %	5.00	2.265	2.688	0.381	1.025

ET-RR-1 reactor has negative temperature coefficient of reactivity. The estimated value of temperature coefficient of reactivity deduced by the vendor[8] is = -0.24×10^{-4} $\Delta k/k$ per °C for coolant temperature above 25°C, experimental measurement gives a value of -0.32×10^{-4} $\Delta k/k/°C$.

Signalling System

This system consists of three circuits; namely emergency shut-down (scram) with red light signals, warning with yellow light signals, and normal operation with white light signals. The output of the monitor unit C407 of each measuring channel(31 channels), is connected to light and acoustic signalling system. It has the following characteristics:

a) Light signal is generated in an abnormal measured value. It remains unless its default is recovered.

b) An acoustic signal is generated for each light signal

c) Three signals are directly connected to the emergency circuit to scram the reactor. They are :

- 20% pressure reduction in primary coolant circuit
- 10% flow reduction in primary coolant circuit
- 20% reactor power increase

Emergency Shut-down Signals

It gives scram and red light signals when one of the following actions takes place:

1- The reactor period is less than 15 seconds
2- Two out of three of period-meters (long channels) are not operating
3- Two out of three of log-channels power violation
4- Two out of three of linear channels exceed 120% of nominal power
5- Two out of three of linear channels are not operating
6- The automatic control rod reaches its lowest position during operation in automatic mode
7- The flow rate in the primary circuit is decreased by 10% of its nominal value
8- Water level in the central tank is less than 5 meters.
9- The temperature of the primary circuit exceeds 50°C
10- primary circuit pressure is reduced by 10% of its nominal value
11- Loss of 220V, AC, that supplies motors of primary, secondary, ventilation, and nuclear systems
12- Loss of 110V Dc, that supplies the servo-drives of the control rods
13- Loss of 48V Dc, that supplies the motors of the safety rods
14- Loss of the function of the CSLS
15- Manual scram (push button)

Warning Signals

It gives warning and yellow flash light when one of the following actions takes place:

- Leak at any pump
- The period meter indicates 30 seconds or less
- The power increases by 10% of the required power
- The period meter indicates 30 seconds or less
- Fail of 127 volt power supply
- Temperature increase of pump bearings more than 45°C
- Flow reduction in primary circuit
- Flow level or rarefaction reduction in dearator
- Pressure reduction in primary or secondary circuit
- Temperature difference increase in primary and secondary circuit
- Fail of automatic control rod
- Fail of one of the two period meters
- Decrease in pumping room rarefaction
- Air activity increases
- Reduction in hot cell rarefaction
- Opening of one of the horizontal experimental channel
- Level reduction in the reactor central tank
- Rarefaction reduction under or above reactor
- Fail of the safety channel ionization chambers power supply
- Horizontal channel open
- Warning signals from CSLS

- Fan off
- Safety rods no pullable
- Manual rods no pullable

Normal Operation Light Signals

It includes the following signals :

- Signals indicating lowest and highest position of each control rod when the rod arrives that position
- It gives signals about the position of the thermal column inside or outside
- It displays the state of the emergency circuit before operation and whether it is ready or not

Emergency Circuit

The emergency circuit is the interface between the process parameters measuring instruments and the control rod. In automatic scram, the emergency circuit provides signals to the safety rod drive circuit so that all safety rods are dropped down inside the core within 0.18 seconds. Moreover, it also gives signal to signalling circuit to indicate the fault to the operator. The operation of this circuit depends on programable logic controllers, which is called computerized safety logic system as described in section 3.

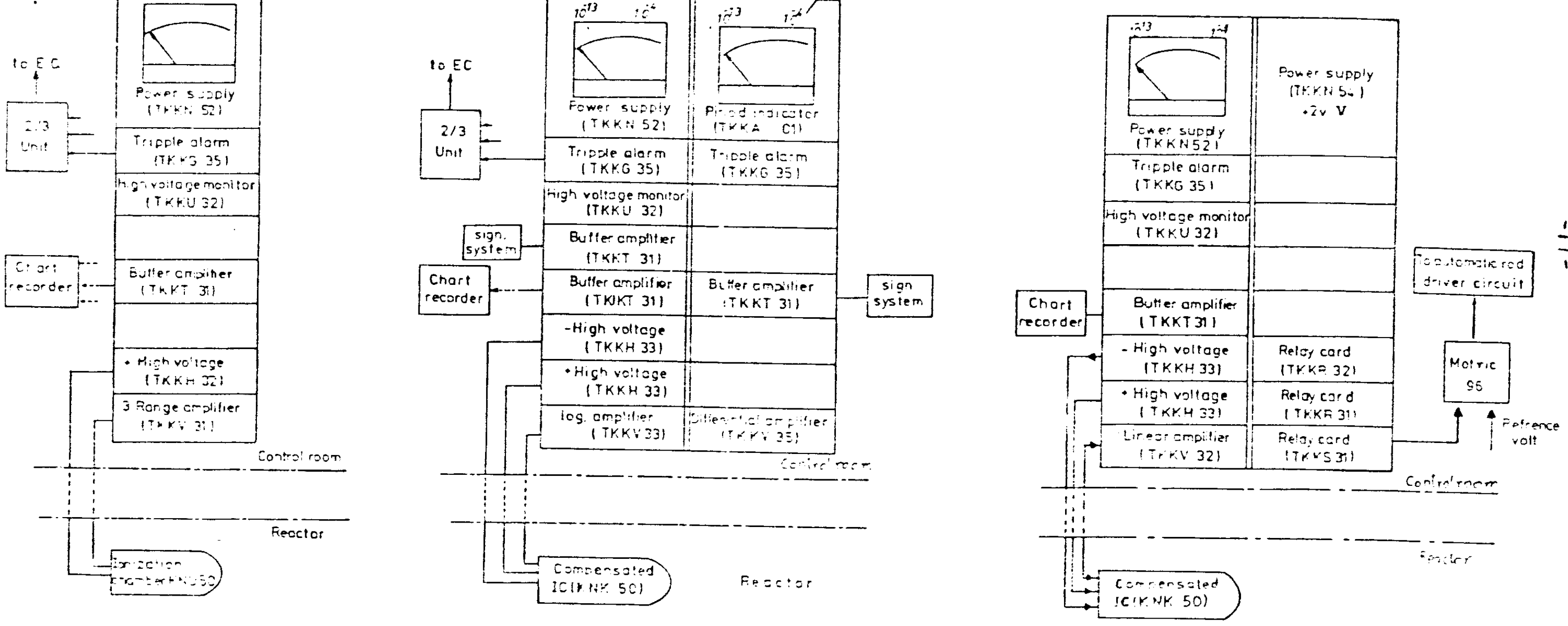
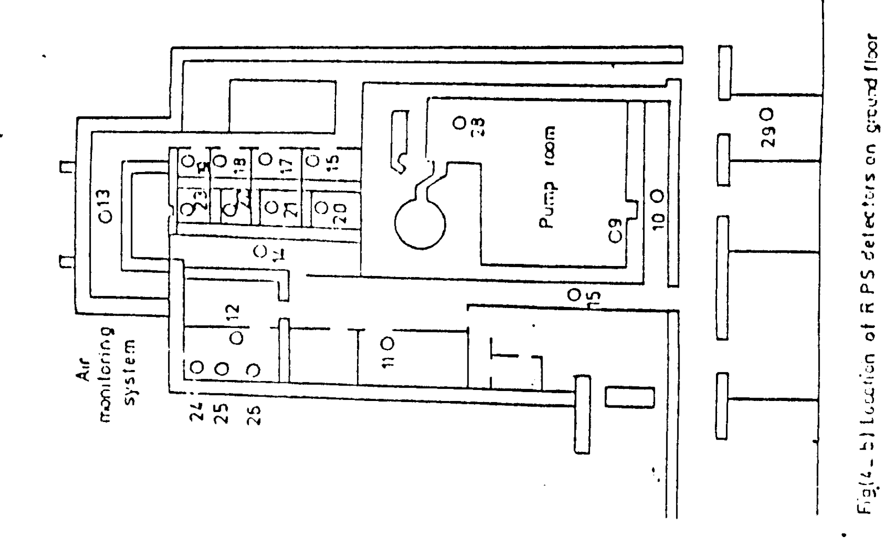


Fig (1) Safety channel (modules in bracket from H&B company

Fig (2) Log. d.c. channel (modules in bracket from H&B company)

Fig (3) Multirange d.c channel (modules in bracket from H&B company)



Fig(4-b) Location of RPS detectors on ground floor

Reactor Hall

DC RPS Centeral room

Fig.(4-a) Location of RPS detectors in 1st Floor

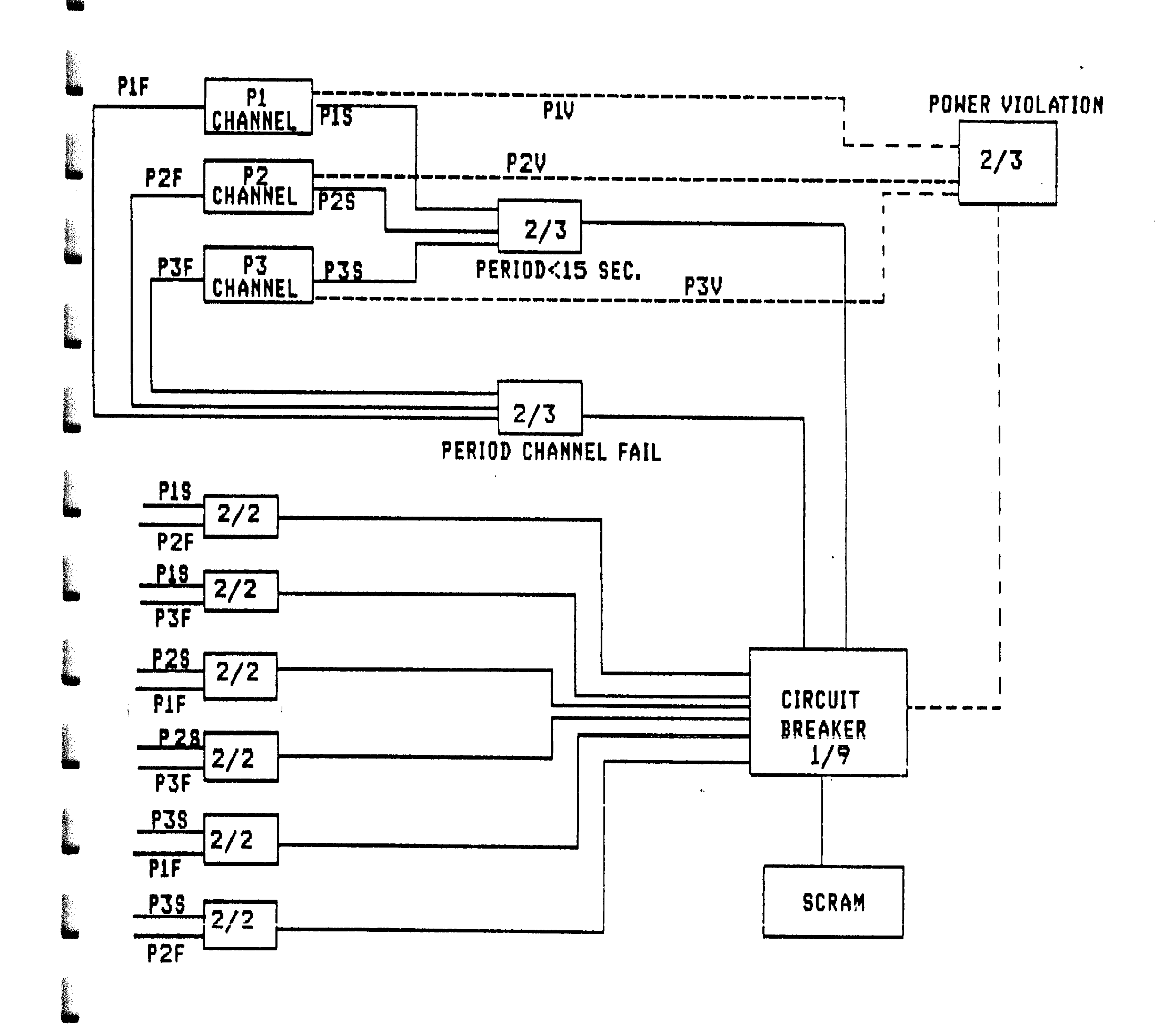
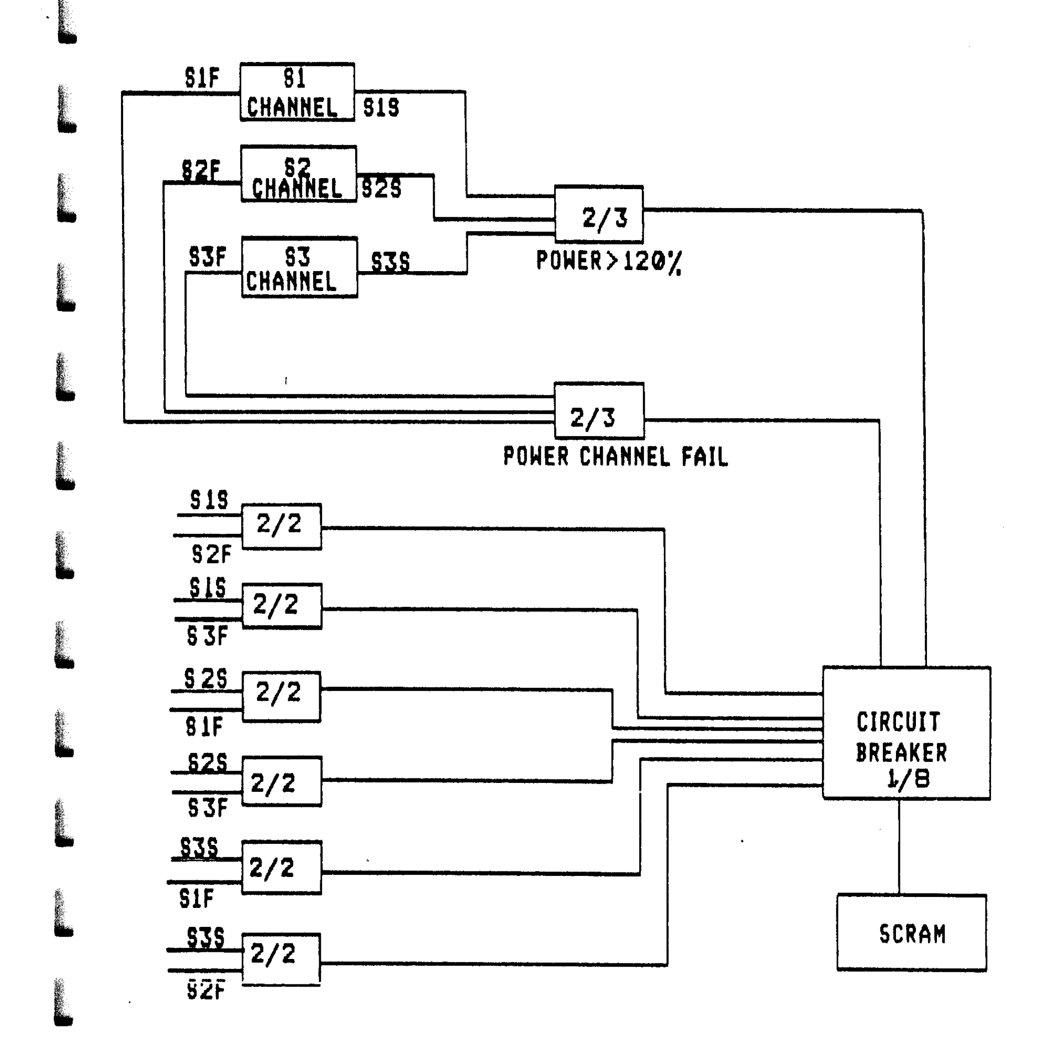
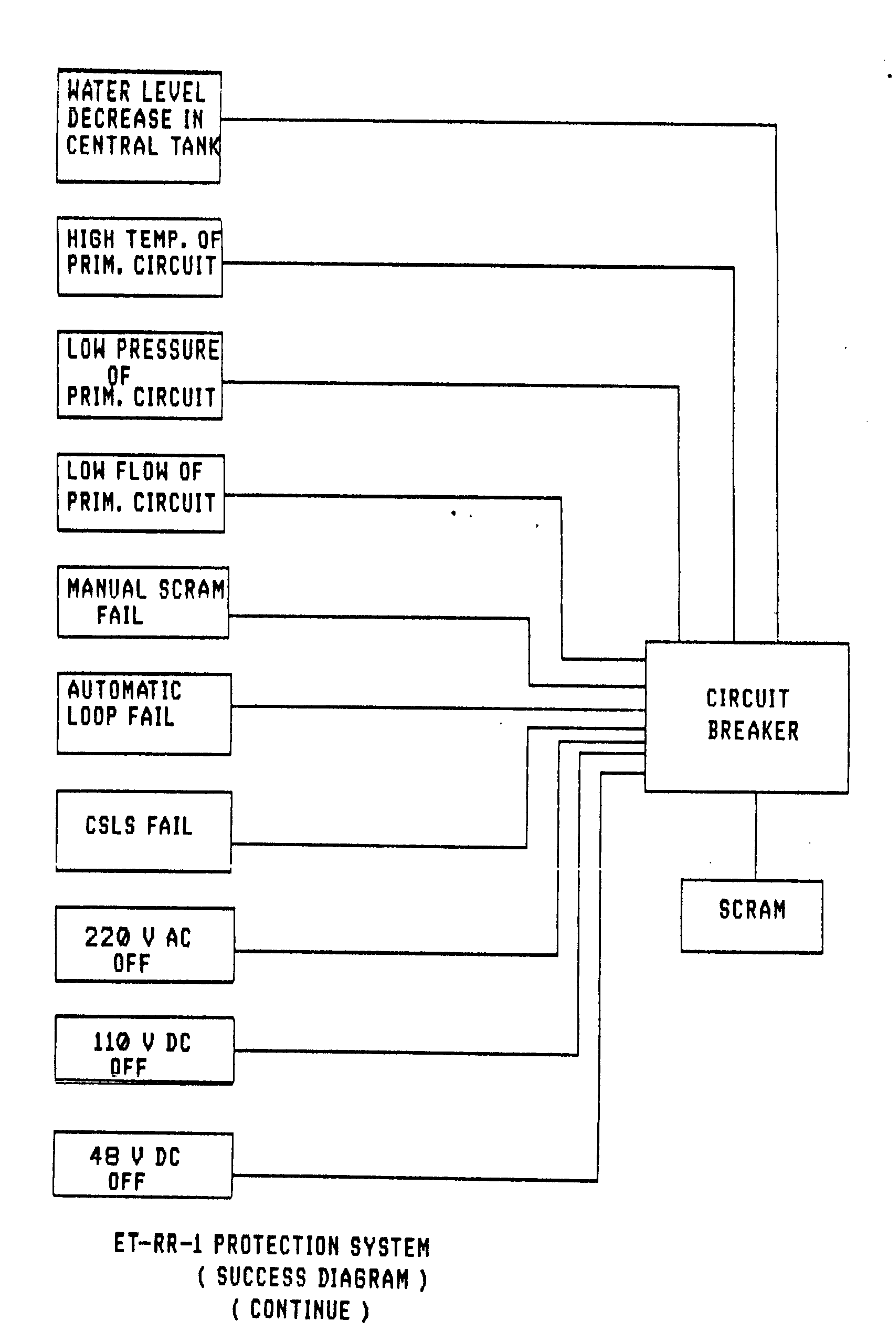


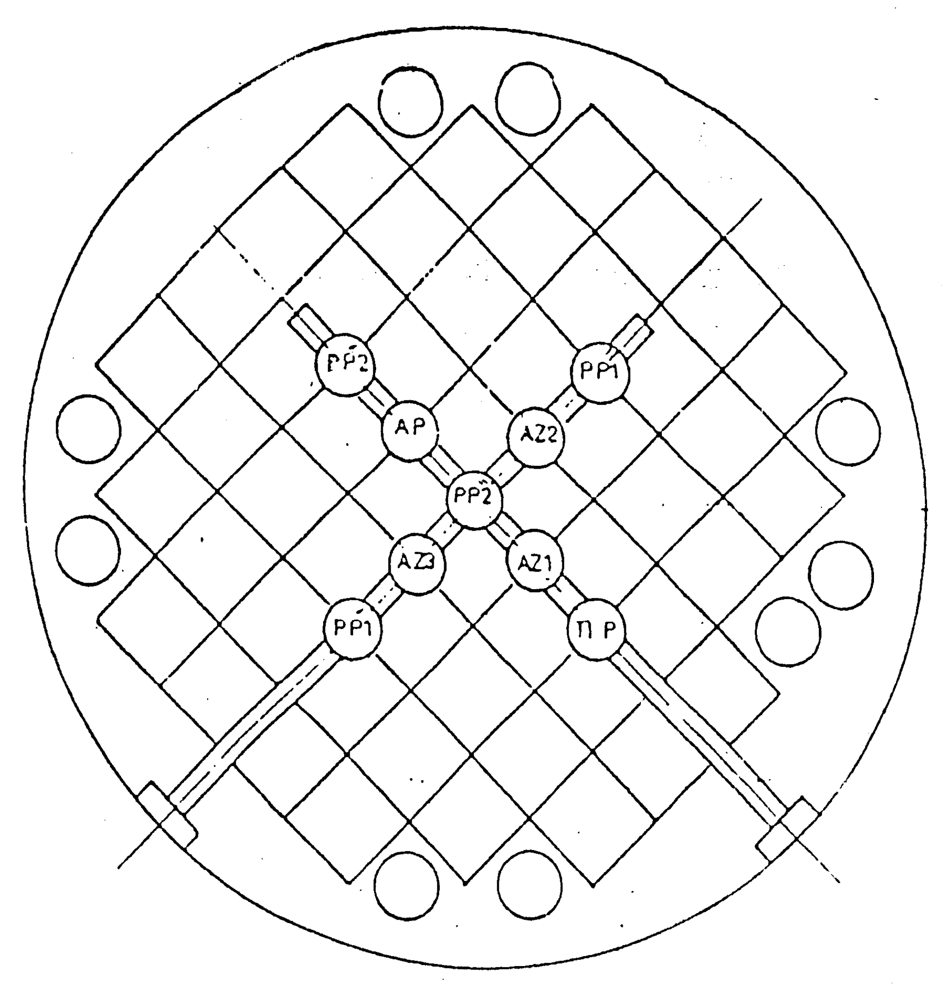
Fig.5 ET-RR-1 REACTOR PROTECTION SYSTEM (SUCCESS DIAGRAM)



ET-RR-1 REACTOR PROTECTION SYSTEM
(SUCCESS DIAGRAM)
(CONTINUE)



ET-RR-1 PROTECTION SYSTEM
(SUCCESS DIAGRAM)
(CONTINUE)



First Shim Manual Rods PP1' and PP1''

Second Shim Manual Rods PP2' and PP2''

Safety Rods AZ1, AZ2 and AZ3

Automatic Regulating Rod AP

Precision Regulating Fine Rod Π P

FIG. (6) DISTRIBUTION OF THE CONTROL RODS IN THE ET-RR-1 REACTOR.

FIG. 7 PROPOSED ARRANGEMENT OF THE INSTRUMENTS (TYP C)

Connection Drawing of the control instrumentation

Fig.No. .8.

SDS
SRP
MAP
AUT.MOD

MULTIPLI-CATION OF THE CONTROLL SIGNALS

SDS
SRP
MAP
AUT.MOD

48 VDC
110 VDC
127 VAC

POWER CONTROLL

48 VDC
110 VDC
127 VAC

PUSH BUTTONS
END POSITIONS

SR1
SR2
SR3
SAFETY ROD CONTROLL PANELS

POWER SUPPLYS TO SERVO MOTORS
END POSITIONS

PUSH BUTTONS
END POSITIONS
ROD POSITION INDICATORS
AUT. CONTROLL SUPPLY

MR1
MR2
MR3
I/N PE. CONT-ROLL
AUT
MANUAL & AUT. ROD CONTROLL PANELS

110 VDC
127 VAC

POWER SUPPLYS TO SERVO MOTORS
END POSITIONS
POSITION INDICATOR

Fig.No. 9
Block Diagram of the RDCS

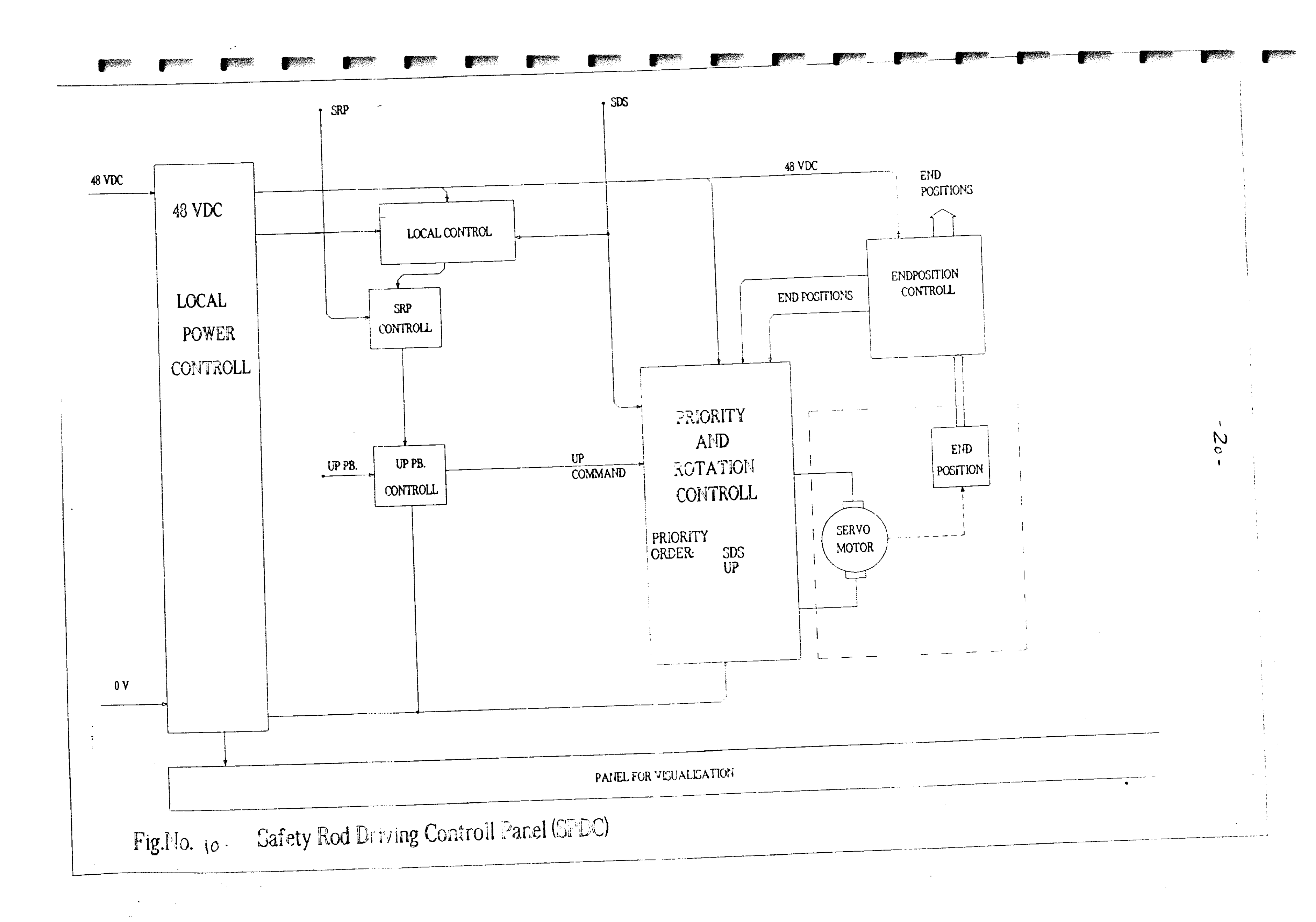


Fig.No. 10. Safety Rod Driving Controll Panel (SRDC)

Fig.No. n · Manual Rod Driving Controll Panel (MRDC)

110 VDC
110 VDC
LOCAL
POWER
CONTROLL
0 V
I/N
PB.
CONTROLL
MAP
AUT. MOD
SDS
AUT.
REGULATION
SUPPLY
110 VDC
END
POSITIONS
127 VAC
127 VAC
LOCAL
POWER
CONTROLL
LOCAL CONTROL
DC/DC
CONVERTER
U<110 VDC
$U_{controll}$
END POSITIONS
ENDPOSITION
CONTROLL
MAP
CONTROLL
DOWN PB.
I/N PB.
CONTROLL
DOWN PB.
CONTROLL
DOWN
COMMAND
PRIORITY
AND
ROTATION
CONTROLL
PRIORITY
ORDER
SDS
AUT
DOWN
UP
SERVO
MOTOR
END
POSITION
ROD
POSITION
ROD
POSITION
INDICATION
TO
OPERATOR
0 V
UP PB.
UP PB.
CONTROLL
UP
COMMAND
PANEL FOR VISUALISATION

Fig No. 12. Automatic Rod Driving Controll Panel (ARDC)

REFERENCES

1. Code on the Safety of Nuclear Research Reactors Operation, Safety Series No. 35-S2, IAEA, Vienna, 1992.

2. Documents of (ET-RR-1), Reactors Department, AEA, Egypt.

3. M.A. Sultan et al , Renewal of Instrumentations for (ET-RR-1), IAEA, SM- 310/36

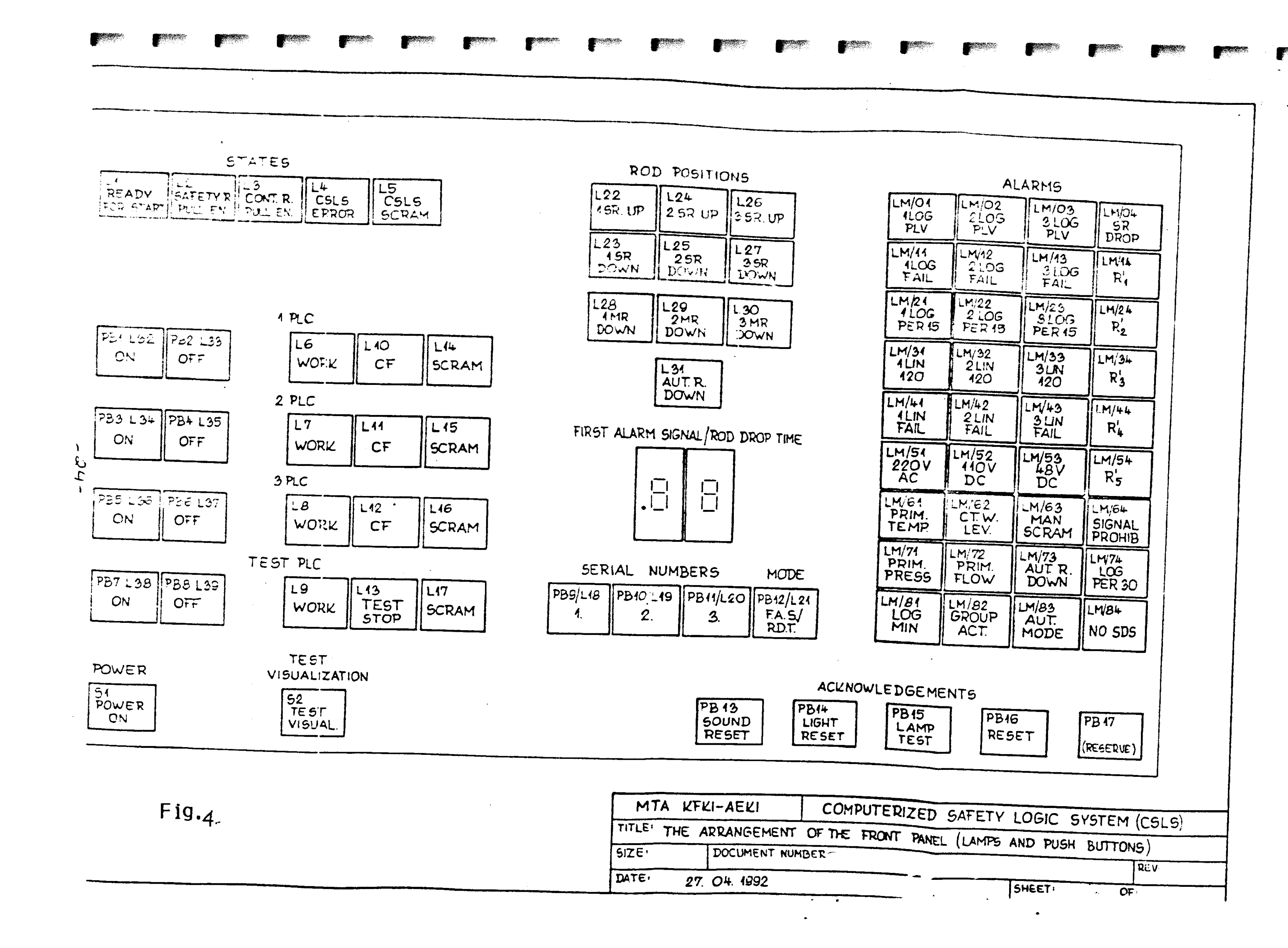

STATES
READY FOR START
SAFETY R. PULL EN.
L3 CONT. R. PULL EN.
L4 CSLS ERROR
L5 CSLS SCRAM
ROD POSITIONS
L22 1SR. UP
L24 2 SR UP
L26 3 SR. UP
L23 1SR DOWN
L25 2SR DOWN
L27 3SR DOWN
L28 1MR DOWN
L29 2MR DOWN
L30 3MR DOWN
L31 AUT. R. DOWN
ALARMS
LM/01 1LOG PLV
LM/02 2LOG PLV
LM/03 3LOG PLV
LM/04 SR DROP
LM/11 1LOG FAIL
LM/12 2LOG FAIL
LM/13 3LOG FAIL
LM/14 R'1
LM/21 1LOG PER15
LM/22 2LOG PER15
LM/23 3LOG PER15
LM/24 R'2
LM/31 1LIN 120
LM/32 2LIN 120
LM/33 3LIN 120
LM/34 R'3
LM/41 1LIN FAIL
LM/42 2LIN FAIL
LM/43 3LIN FAIL
LM/44 R'4
LM/51 220V AC
LM/52 110V DC
LM/53 48V DC
LM/54 R'5
LM/61 PRIM. TEMP.
LM/62 CT.W. LEV.
LM/63 MAN SCRAM
LM/64 SIGNAL PROHIB
LM/71 PRIM. PRESS
LM/72 PRIM. FLOW
LM/73 AUT. R. DOWN
LM/74 LOG PER 30
LM/81 LOG MIN
LM/82 GROUP ACT.
LM/83 AUT. MODE
LM/84 NO SDS
PB1 L32 ON
PB2 L33 OFF
PB3 L34 ON
PB4 L35 OFF
PB5 L36 ON
PB6 L37 OFF
PB7 L38 ON
PB8 L39 OFF
1 PLC
L6 WORK
L40 CF
L14 SCRAM
2 PLC
L7 WORK
L41 CF
L15 SCRAM
3 PLC
L8 WORK
L42 CF
L16 SCRAM
TEST PLC
L9 WORK
L13 TEST STOP
L17 SCRAM
FIRST ALARM SIGNAL/ROD DROP TIME
SERIAL NUMBERS
PB9/L18 1.
PB10/L19 2.
PB11/L20 3.
MODE
PB12/L21 F.A.S/ RDT.
POWER
S1 POWER ON
TEST VISUALIZATION
S2 TEST VISUAL.
ACKNOWLEDGEMENTS
PB13 SOUND RESET
PB14 LIGHT RESET
PB15 LAMP TEST
PB16 RESET
PB17 (RESERVE)
MTA KFKI-AEKI
COMPUTERIZED SAFETY LOGIC SYSTEM (CSLS)
TITLE: THE ARRANGEMENT OF THE FRONT PANEL (LAMPS AND PUSH BUTTONS)
SIZE:
DOCUMENT NUMBER
REV
DATE: 27. 04. 1992
SHEET:
OF:

Fig.4.

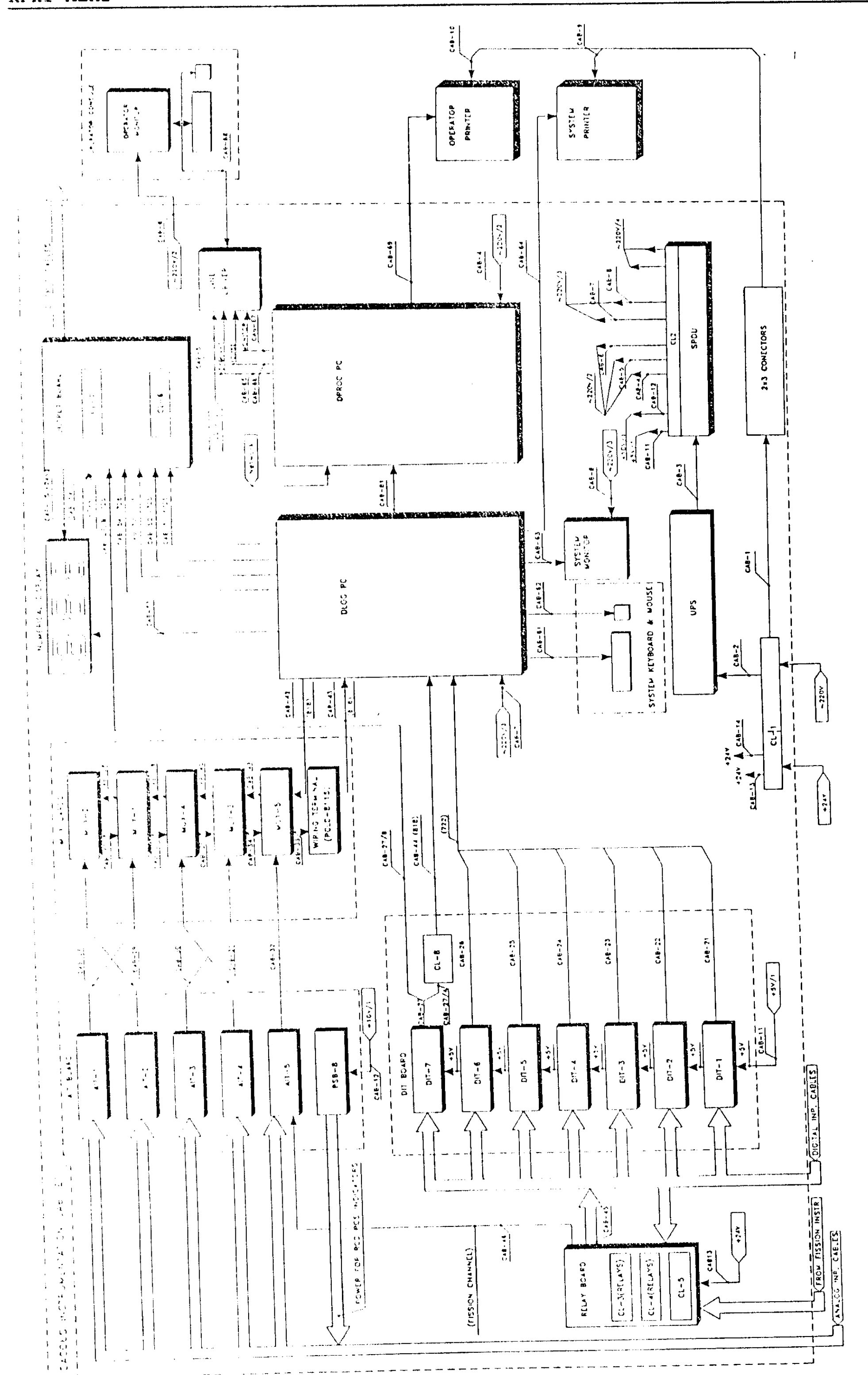


Fig. 5. BLOCK SCHEME OF THE DACQUS (Hardware Configuration)

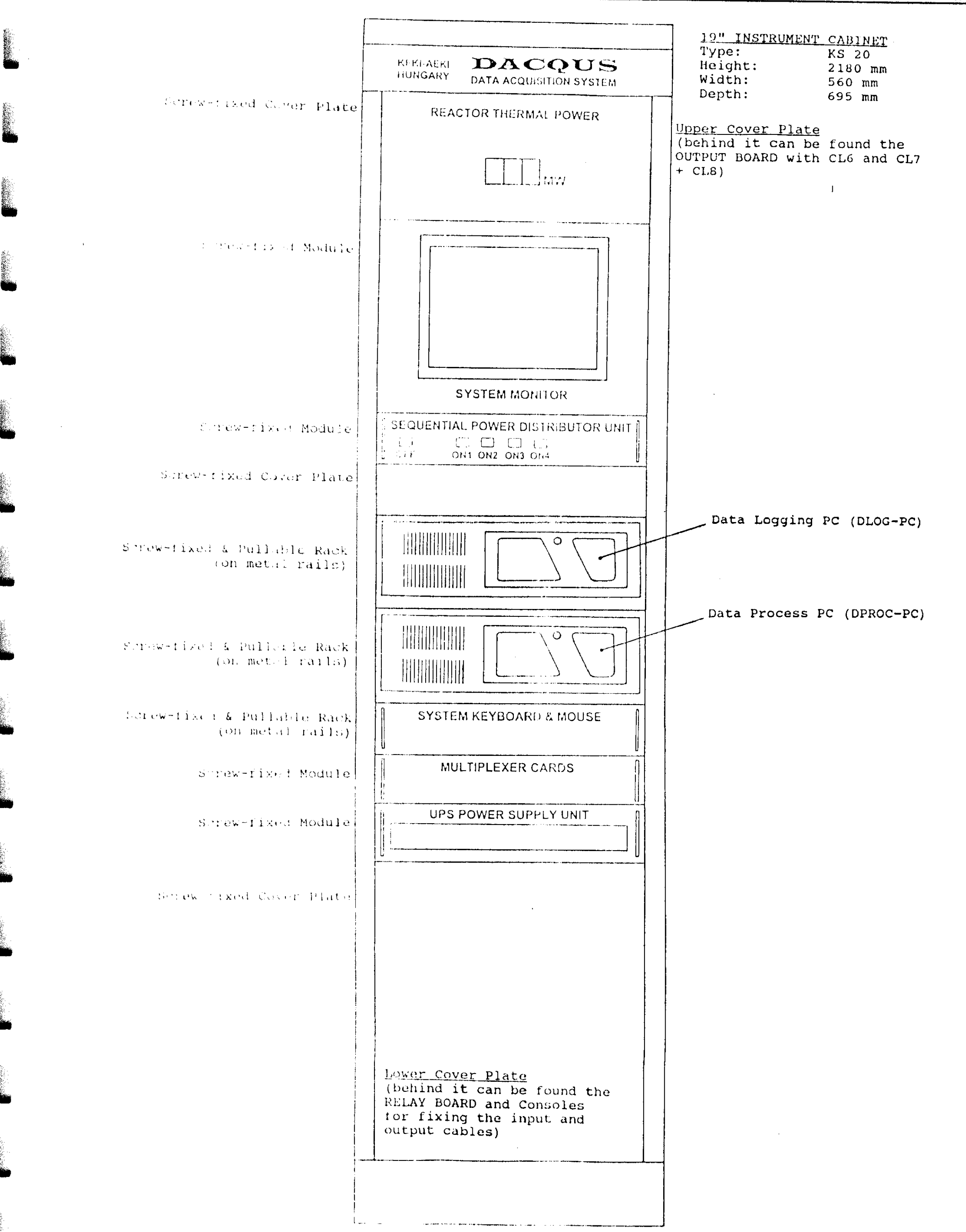


Fig.1.1. CUPBOARD FRONT-VIEW

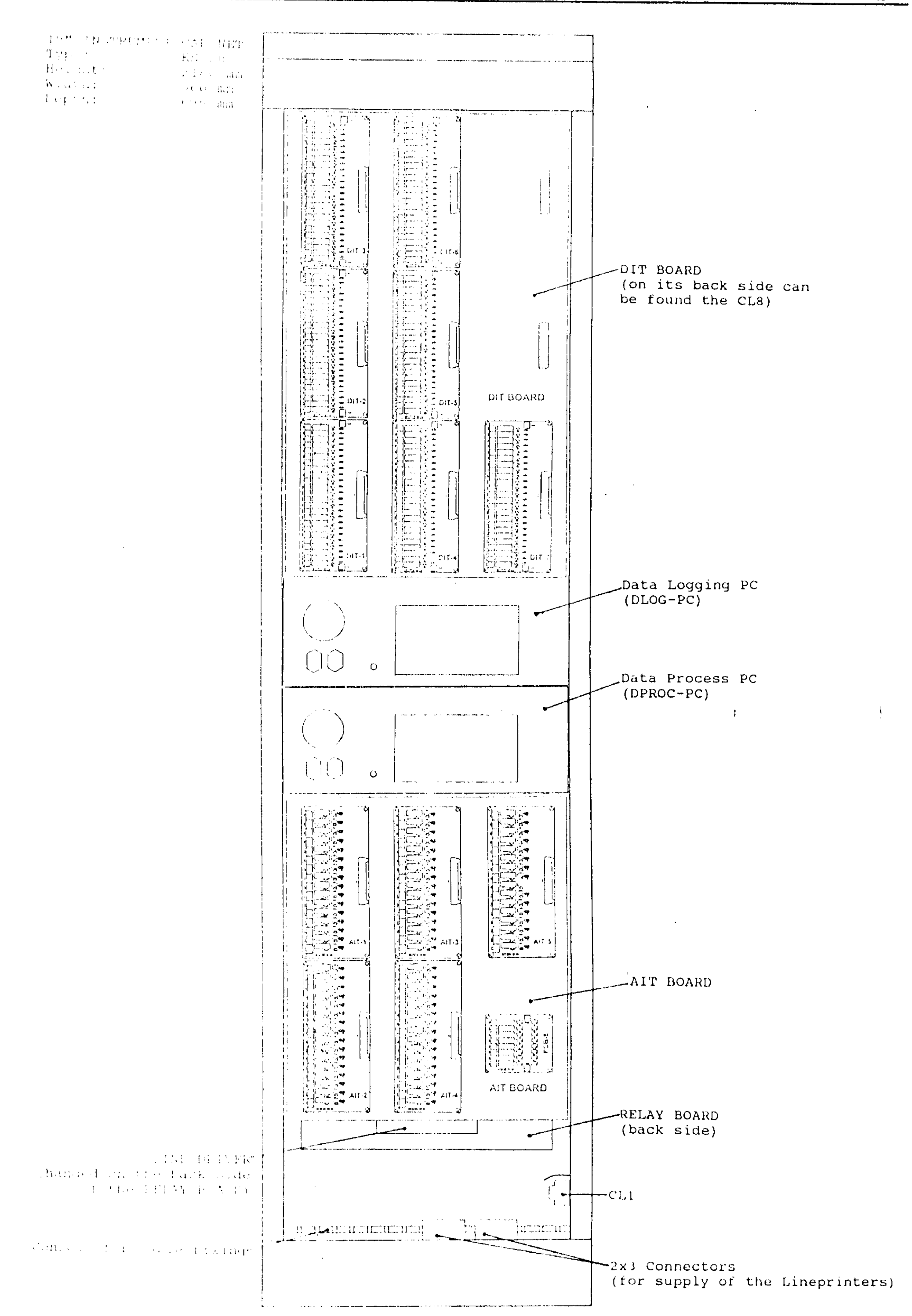
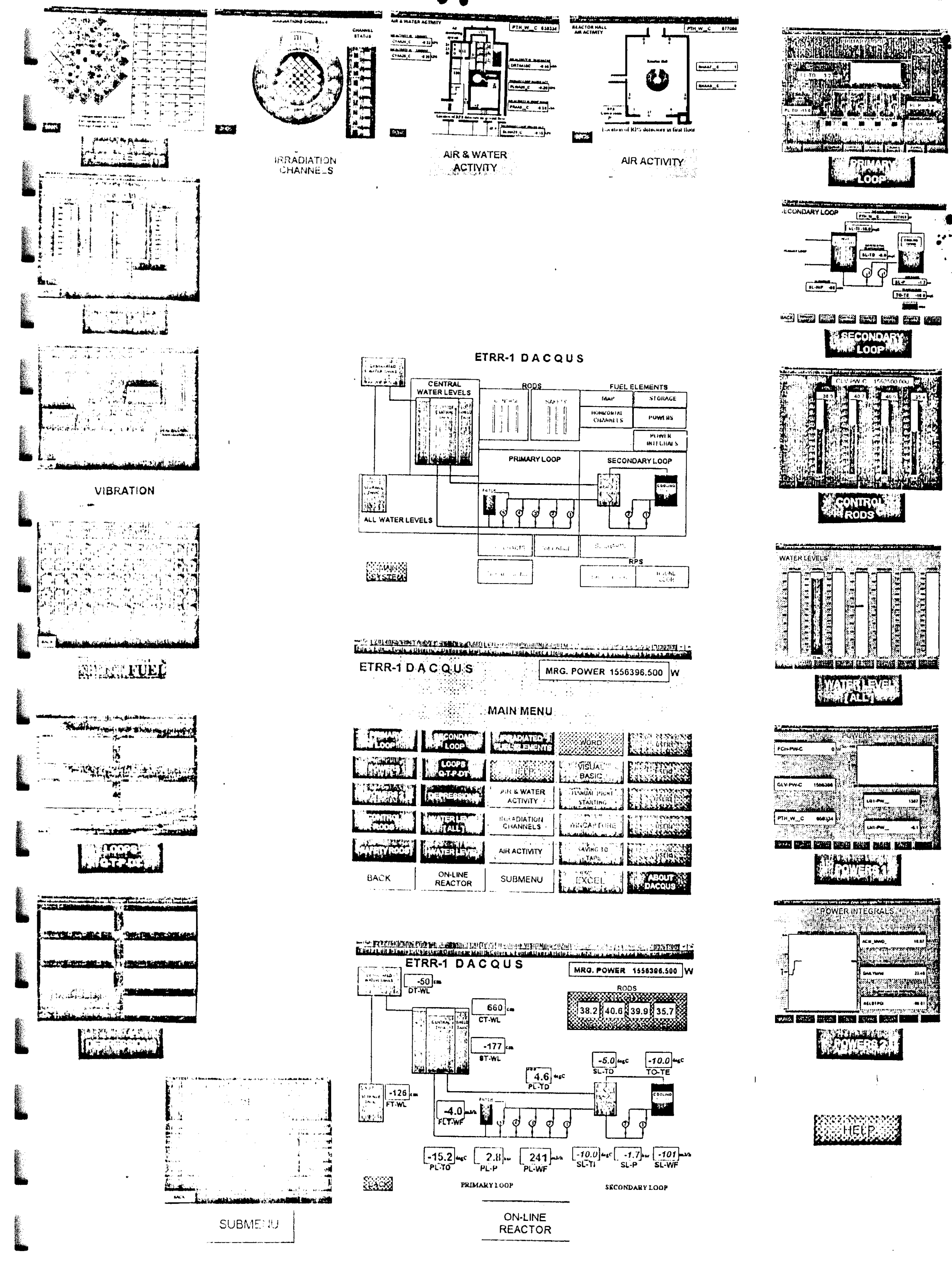


Fig.1.2. CUPBOARD REAR-VIEW


IRRADIATION CHANNELS
AIR & WATER ACTIVITY
AIR ACTIVITY
PRIMARY LOOP
SECONDARY LOOP
VIBRATION
CONTROL RODS
WATER LEVELS
WATER LEVEL (ALL)
FUEL
LOOPS Q-T-P-DT
DEPRESSIONS
POWERS 1
POWERS
POWER INTEGRALS
POWERS 2
HELP
ETRR-1 DACQUS
CENTRAL WATER LEVELS
RODS
FUEL ELEMENTS
MAP
STORAGE
HORIZONTAL CHANNELS
POWERS
POWER INTEGRALS
ALL WATER LEVELS
RPS
MAIN MENU
MRG. POWER 1556396.500 W
WORD
VISUAL BASIC
AIR & WATER ACTIVITY
IRRADIATION CHANNELS
AIR ACTIVITY
SAVING TO TAPE
EXCEL
ABOUT DACQUS
BACK
ON-LINE REACTOR
SUBMENU
-50 DT-WL
660 CT-WL
-177 BT-WL
-126 FT-WL
38.2 40.6 39.9 35.7
4.6 PL-TD
-5.0 SL-TO
-10.0 TO-TE
-4.0 FLT-WF
-15.2 PL-TO
2.8 PL-P
241 PL-WF
-10.0 SL-TI
-1.7 SL-P
-101 SL-WF
PRIMARY LOOP
SECONDARY LOOP
SUBMENU
ON-LINE REACTOR

Leg. 5.3

COMPUTERS IN MEASUREMENTS AND DATA PROCESSING

BY

M.K.SHAAT

INTRODUCTION

Recently, the Atomic Energy Authority of Egypt (nuclear reactor department) introduced the computer technique in the reactor protection system of its first research reactor (ET-RR-1) build since 1960 and located in Inshass area.. This technique have several important advantages such as :

- very compact construction,
- easy change of the logic,
- simple self testing methods,
- extreme high reliability, and
- event logging after shut-down

Due to the above advantages, the Programmable Logic Control (PLC) based reactor protection systems have been developed in different countries. The computerized safety logic system (CSLS) is based on EBERLE made PLCs. the reactor department used the experience gained from the design and realization of similar CSLS of the KFKI research reactor in Hungary. The CSLS system was installed and tested in May 1992.

In the following a description of the CSLS system was given and after that the reliability calculations of the scram system of ET-RR-1 reactor.

DESCRIPTION OF THE CSLS[1,2]

Hardware Building:

The CSLS has two main parts:

a) The control panel with the lamps and push buttons. This unit is fastened with screws to the main frame of the operator console.

b) The hardware of the CSLS together with the four PLCs. This unit is mounted on a self-supporting frame.

Between the two parts there are about 3 meters long cable connections. An additional part of the CSLS is the connection box which is served to connect the input signals from the instrumentation of the reactor to the CSLS and through this box the CSLS gives out all output signals. The contact between CSLS and connection box is made by four cables. These cables are drawn in a flexible metal tube. The main units are the following :
- Opto coupler unit (OPC), - PLC1,PLC2,PLC3 programmable logic controllers,
- PLCT check unit (test PLC), - Voter logics and additional relays,
- Lamp-matrix for signal visualization, - Numeric display, - Control and signaling lamps, - Power supply, - Connection box.

Software Program Modules of CSLS:

The software of CSLS is divided into two parts. The first one deals with the programming of the three PLCs. The second part deals with the programming of TEST PLC. A list of these programs are:
1- The scram program, 2- The interlock program 3- Program of signalization
4- Program to store the first alarm signal, 5- Program to measure the safety rod drop time,
6- Program to check the WORK state of PLCs 7- The cyclic test program, and
8- Logic and hardware self check program

TECHNICAL PARAMETERS

The technical prameters consist of operating, electrical, and mechanical prameters.

Operating Prameters:

It includes : - Base functions, - Additional functions, - Number of input signals, - Number of output signals, -Structure of the system, - Number of PLCs, which are display, control, and cabling.

Electrical Parameters:

It includes: - PLC type, - CSLS power, PLC memory type, - Relays types, - Input signals voltage, and - Output signals.

Mechanical Parameters:

It includes: - Operating temperature, - CSLS dimension, Connection box, and - CSLS weight.

Input Signals List:

The total number of input signals are 64. The used signals are 54 and the rest (10) are reserved. The input signals can be classified as:
- Signals of the triplicated nuclear measurements, their number is 18 (15 in use and 3 reserved).
- Independent alarm signals. Their number is 7 (6 in use, and 1 reserved).
- Alarm signals depending on the reactor operating mode. Their number is 3 (2 used, and 1 reserved).
- Operating mode signals. Their number is 2 (2 used).
- Rod position signals. Their number is 10 (10 used).
- Interlock signals. Their number is 15 (11 used, and 4 reserved).

These signals are used for :
- Safety rod interlock. Their number is 6 (5 used, and 1 reserved).
-Shim and automatic rod interlock. Their number is 9 (6 used, and 3 reserved).
- Signals from push buttons for the signalization Their number is 9 (8 used, and 1 reserved).

Note that the logic definition for the used inputs is logic 0 (means opened contacts in the inputs), and the reserved inputs must be short circuited at the inputs (logic 1).

Output Signal List :

The output signals can be classified as:
- Function outputs, - Signaling outputs, and - Signaling for DAQUS.
Note that :
- Each signal output has got only one switched contact.
- Signaling for DAQUS are built only to the CSLS connection line.

List of Logic Functions:

The logic functions are written for 3 functional outputs only. The meaning of the other outputs can be followed from the state of the mentioned 3 outputs. The three functional outputs are: - Shut-down, - Safety rod pullable(SRP), and - shim (Manual) and (Automatic) rod Pullable (MAP).

OPERATING OF CSLS

This part includes the following:
- Starting of CSLS, - Continous operating of CSLS, - Safety rod drop-time measurement,
- Safety rod calibration, - Sorting and display of first alarm signal, - Light and sound signaling,
- Stopping the CSLS, and - Interpretation of CSLS scram function.

CONTROL AND MAINTENANCE OF CSLS

Control and maintenance of the CSLS always has to be done by, at least, two persons.

Controls:

The aim of control is to establish the working order of CSLS. Control has to be done by proper use of signals and cuff-links in TEST mode. The CSLS is a part of the whole reactor measuring, control, safety and alarm system. So control of CSLS must be expanded from signal sources to executive organs.

The following control steps must be done according to the determined periods of time, but these control stages must be repeated each time if CSLS itself or its connecting parts have been repaired. In this case, the user determines the control steps depending on the events that happened in the reactor's operation. The general points-of-view for control are:
- Visual inspection, - Measuring of insulation resistance of signal cables,
- Measuring of power consumption of CSLS
- Control of input signals on: - opto coupler, - input of PLC, and - lamp matrix field
- Control of outputs on: - PLC outputs, -control of VL1,VL2, and VL3 logic, and
- Control of alarm and safety sound signal outputs.

The control steps that should be done are:
1- Before the starting up of the reactor, working in normal operation cycle.
2- During operation at changes of shifts.
3- During the annual maintenance or at least once a year.
4- After every 5 years.

Maintenance:

Maintenance of CSLS. means execution of regular checks, including control and refreshing of software. CSLS does not contain components requiring maintenance. The following steps must should be done:
1- Periodical changes of components, 2- Failure exploration, and 3- Repairs(hardware and software)

SAFETY FEATURES OF CSLS

The main safety features of the CSLS are:
1- Fail-safe system, 2- Redundancy 2/3 coincidence system,
3- Increased availability with high reliability which means that for single failure, the system safety operates in 1/2 structure,
4- Easy possibility for the functional test, 5- Continuous system check,
6- Galvanic isolation from the environment, 7- To store the first alarm signal,
8- To measure the safety rod drop time, 9- Interlocking the safety and manual rods,
10- Reactor period must be larger than 30 seconds,
11- Increase in reactor power must be less than 12% of operating power,
12- Sound and light alarm signals, 13- Easy in control and maintenance,
14- Easy change of logic functions, 15- Event logging after shutdown,
16- Self-locking relay. It means that if the relay drops for SDS, it can only switch-on again if the READY to START state exists and the operator pushes the START UP ENABLE button, and
17- Double failure, which means that the failure of two PCLs or the connected circuit causes the shutdown of the reactor.

SCRAM SYSTEM RELIABILITY CALCULATIONS

The failure probability (unavailability) of the scram system of ET-RR-1 was studied. The fault tree of the system was constructed, and the reliability data (failure rates) of the system components were taken from the specific operation data of ET-RR-1 reactor. Also, generic data for general components of the system were used in case of that the specific one are not available. The failure probability (unavailability) of the scram system was calculated using the PC-code PSAPACK[3]. The results were analyzed and discussed to show the marginal operational safety of the reactor.

System Description:

The function of the scram system as producing (automatically or even manually) a fast decrease of power in the reactor when the integrity of the physical barriers containing the fission products and the fuel elements are threatened. The protection system consists of the following elements:
1- Sensors which monitor permanently the development of the physical parameters representative of the functioning of the reactor, either in normal or abnormal situations.
2- Equipments transforming the signal sent by sensors into measurable electric currents (amplifiers,etc...).
3- Comparators actuating a signal indicating that the physical parameters measured have exceeded the set value (threshold value).
4- Computerized Safety Logic System (CSLS), which contains logic circuits grouping and processing the signal from the comparators and giving the scram signal.
5- equipments called circuit breakers which from the scram signal directly actuate the servo-drive loop (emergency circuit) to insert the control rods into the reactor core by gravity.
6- Control rods whose function is to terminate the fission reaction in the core by fast absorption of thermal neutrons.

Actuation of the Scram System :

The scram system is designed in order to respond to a long list of initiating events, which are considered as potentially leading to abnormal consequences or to accidental conditions. Each initiating event characterized by some measurable parameters which if they

exceed certain safety limits it will actuate a shutdown signal to the scram system which induce the control rods to fall down into the reactor core. The control rod drive mechanical reliability will not be part of the study as quantified values, but studied qualitatively.

Flow of Scram Signal:

Figure (1) shows the flow of each scram signal from left to right. Each signal measured by a sensor. The signal will be transferred to a measurable electric current by the amplifier. This signal will be compared by a reference safe limit value. If this signal exceeds the safe limit value, the signal will be proceeded into the CSLS and the scram order will be given to the control rod drives of the safety rods.

Success Diagram:

Figure (2) is a schematic, from left to right, shows the sequence resulting from the scram signal followed by an effective scram. There are 27 signals, one of them should give a scram order to the CSLS which in turn will disconnect the current from the control rod drive circuit and the safety rods will be inserted inside the reactor by gravity force and spring force.

Assumptions:

1- The manual emergency shut down was not included in the fault tree, since it intervenes directly on the logical system.
2- The failure of 220V AC power supply was not included in the analysis, because it acts as a fail-safe signal which will scram the system directly.
3- Because the CSLS is a fail-safe system, as a result of the safety feature of the system, the failure of CSLS was not included in the fault tree.

DATA ACQUISITION SYSTEM (DACQUS)

1. GENERAL INFORMATION

1.1. Introduction

Equipment of the programmable technology can provide new capabilities to protection, control, data acquisition etc. systems of nuclear research reactor. The Atomic Energy Research Institute, Budapest encourages the application of advanced technology such as Programmable Logic Controllers (PLC) and Computerized Data Acquisition Systems in the operation of nuclear research reactors if such advanced equipment serves to enhanced safety and more advantages information for the operator.

This Data Acquisition System, named DACQUS was constructed directly to the ETRR-1 Research Reactor in Inshass by the Tender Prescriptions of the Atomic Energy Authority, Nuclear Research Centre, Egypt - Cairo (Tender No. 15 Due: 25/12/1994). In the system design the Vendor (KFKI, Atomic Energy Research Institute, H-1525 Budapest, Pf. 49., Hungary) used the experience that gained from the design, realization and operation of the similar Data Acquisition System of the KFKI-AEKI research reactor. The system conception of the DACQUS is fitted into the special systems installed to the ETRR-1 Reactor by the Vendor the last years. Due to this design conception the Safety Logic System (CSLS) with the new operator console and the Signalling System (SYS) are based on the same system design installed in our reactor. These systems (inclusive the data acquisition and rod control systems) are constructed in a uniform conception, so they are built on each other in a certain degree.

Main parameters of the DACQUS and all set-up conditions (inclusive the signal lists, signal conversion factors, schematic drawing, the main features of the signal handling etc.) were accepted by the User (Reactor Department, Nuclear Research Centre, Inshass, Egypt).

1.2. Meaning of the Data Acquisition

The DACQUS provides the data logging of every important measured and displayed analogue and digital signals of the Research Reactor. The collected data are available for the operators in

form of schematics, charts and graphs; and they are archived. The DACQUS/M is a very important aid for the reactor operators and provides data for the reactor users.

The DACQUS does the supervision of the reactor instrumentation, records the events, the activity of the operators, but does not replace any task of the built in instrumentation of the reactor. The DACQUS is a simply data logging system only.

▪ **VERY IMPORTANT WARNING**

The DACQUS improves the reactor safety and gives powerful help to operator but *does not replace the built in reactor instrumentation!*

3. FEATURES

Having sum the advantageous features of the DACQUS have to be mentioned two base features regarding the hardware and software:

(1) Hardware
Although the DACQUS is constructed for special reactor using, the hardware setting up of the system is based on PC standard computer modules and standard PC-LAB cards, which can be available in any special computer dealers dealing with process control and instrumentation.

(2) Software
Menu-driven software with icon-based programs. This software solutions of the system are mostly based on standard software packages, which give flexible possibility (menu-driven) for the user in certain frames.

- **Further features**

(3) The collected data are available for the operators in form of schematics, charts and graphs in real time.

(4) The system contains built-in quantitative development possibilities (built in reserve hardware signal inputs), of course, but the standard hardware setting up, besides of the changeability of the cards, provides the qualitative development of the system in the future as well.

(5) The signals of the analog and digital outputs are assembled that, they can give a possibility (with other supplements) to provide an additional (parallel) control of the Safety Logic System and Signalling System like a supervisor system (so called: cover protection).

(6) This computer configuration by the built-in network software gives possibility to attach with further PC-s to the DACQUS as intelligent terminals.

- **Main task of the DACQUS:**

- data collection,
- parameter calculation,
- validation and level control,
- records and log books,
- archivation and storage of the analogue signals for post-mortem analysis purposes,
- operator aid,
- data provision for the reactor users and
- vibration monitoring for main mechanical parts of the primary loop (as input signal).

4. TECHNICAL PARAMETERS

4.1. Operating parameters

Number of input analogue signal:	80 pcs
From these: - current input:	64
- supply input	16
Current input range:	0...20 mA (DC)
Supply input range:	0...10 V (DC)
Number of input digital signals:	168 pcs
Signal type:	independent contacts
Control supply:	5 VDC (from built in power supply) with 3,3 kOhm
For signal multiplications	
- built in copy-rely:	24 pcs
relay type:	24 VDC, with 3 morse contacts
Analogue signal sampling time:	1 sec
Digital signal sampling time:	1 sec
Signal correction:	possible at every analogue signal
Calculated parameters:	22 pcs
System Timing:	TOD – Time of Day for all signal handling
Signalization:	- system monitor - operator monitor - 3 digit numerical display
Control:	- system keyboard - operator keyboard
Analog Log-Files	
– Circular Log-File:	- CH: 24 pcs by hours closing - CM: 6 pcs by 10 minutes closing
– Archivation Log-Files:	- LH: Once on hours of day logging (by hours) - LM: Once of minutes of day logging (by minutes) - ACMWD: Accumulated thermal MWDay (by sec)
– Start/Stop Log-Files:	- MM: Manual mode logging

	(by Sec) - SD: Shut-Down logging (by sec)
Digital signal recording:	- ALARM Log-File (time and direction of the polarity changing as event record) - few of them as analog signal in the MM and SD Log-Files (duplicated recording)
Storage capacity:	4 x 500 MByte winchester
Signal archivation:	- streamer tape (120, MByte) - 1.44 MB disc
Hardcopy:	- Log book by Hours (from LH-log) - Digital signal changing (at events or summarized)
Outputs:	
- Analog output:	6 supply (0..10VDC,max.5 mA)
- Digital output:	16 relay contacts (max. 24 VDC, 250 mA)
Local Network:	ETHERNET max. 15 users
Operator support:	- ....ready-made schematic drawings - data charts - on line and off line trend analysis

4.2. Electrical parameters

Power consumption:	200 V, 50 Hz, 800 VA
Built in UPS capacity:	750 VA for two PCs only without printers
Load time at power cut:	max. 15 minutes
Sequential power distribution:	4 levels (self shut-down)
Inside powers:	- 5 VDC, max 1 A (for digital inputs), - 10 VDC, max 0,5 A (for rod position indicators)
Power for relays and numerical display:	24 VDC, 3 A

4.3. Mechanical parameters

Placed in:	19" Instrument Cabinet with

	back door and removable covering, type KS20
Dimensions - width:	560 mm
- height:	2180 mm
- depth:	695 mm
Weight:	approx. 150 kp
Operating temperature:	0 ... +55 °C

4.4. Technical parameters of the supported parts

<u>Vibration monitoring part</u>

Sensitivity:	10 pC/m/s^2 ± 15 %
Linear frequency range:	0 ... 2000 Hz
Analog output:	4...20 mA
ALARM 1 output:	2-20 mm/s RMS
ALARM 2 output:	2-60 mm/s RMS
Length of the measuring cable:	10 meters
Local Power Supply:	220 V, 50 Hz, 5 VA/pc

<u>Rod position indicator transmitter</u>

Sensor:	500 Ohm/2 W Helipot
Indicator range:	10 rotations
Gear transmission:	1:1
Position setting:	mechanical zero adjustment
Control supply:	10 VDC

5. DETAILED HARDWARE DESCRIPTION

5.1. Hardware configuration

The hardware drawings and tables connected with the hardware can be found on Supplement No. 1. (our references connected with the hardware will be on numbering of this supplement).

The block scheme of the DACQUS can be seen on Fig. 1. The central part of the system is built in a 19" instrument cabinet as can be seen on Fig. 1.1. and 1.2.

5.2. Power consumption

■ **220 VAC power**

The computer parts of the DACQUS (exept input relays and numerical display) operates from 220 V, 50 Hz, 800 VA. The system has a built in UPS unit with 750 VA loading. For the divide loading at switch on, the system is supplied a so called Sequential Power Distributor (SPD) Unit (Fig.2, 2.1...2.5) which allows to switch on the power supply to the computer parts in the following orders:

1. level: Built in power supply PWR 5 V, and PWR 10V (for signalization).
2. level: DPROC PC and its peripheral units (except the printer)
3. level: DLOG PC and its peripheral units (except the printer),
4. level: reserve.

The SPDU contains ON push buttons controlled self holding relays (one by levels) and one OFF push button. The switch of system of 8the SPDU - because of the self holding solutions of the relays - is self shut down system.

■ **24 VDC power**

For the input multiplication relays and the Numerical Display the DACQUS needs to supply with 24 VDC. The maximum current consumption is 3 A.

5.3. Computers and peripheries

The system configuration consists of two PC-computers built in industrial boxes. One of them serves for the data logging (DLOG-PC), while the other one serves for the data processing (DPROC-PC). The input and output parts are based on PCLAB-cards. The hardware configurations of both PC can be seen on

The Data Logging part can receive 80 analogue and 168 digital signals. The analog input signals should be normalised between ± 10 V by passive signal condition devices. The passive signal conditions are the Analog Input Terminals (AIT) and Multiplexer boards, drawings of which can be found on Fig.6, 6.1,...6.4.and Fig.8, 8.1 and 8.2. The Power Supply Board (PSB-8) belongs to the Analog signal conditions unit (Fig.6.5 and 6.7). The PSB-8 distributes the + 10 VDC control supply for the rod position indicators. The AIT-16 and PSB-8 can be found the ANALOG INPUT TERMINAL (AIT) BOARD (Fig.6.).

The digital signals should be contact signals, while the control supply is 5 V (suitable for TTL logic). The needed passive signal condition are realised by Digital Input Terminals (Fig.7.1...7.3) which are placed on DIGITAL INPUT TERMINAL (DIT) BOARD. The DIT Board arrangement are illustrated by Fig.7. The digital signal condition belongs to the input multiplication relays which are located on RELAY BOARD. Arrangement of the RELAY BOARD and pean assignment of the relays can be found on the Fig.5 and 5.1.

Important periphery of the DACQUS is the OUTPUT BOARD on which are located the Analog outputs (Connection Line 6 – CL6) and Relay output for digital output (CL7). The analogue output signals are in range 0...10 5 V, the digital ones are relay contacts.

An extra built in unit of the DACQUS is the 3-digit Numerical Display which will show the Thermal Power of the reactor in MWs. The input signal of the unit is generated RS232 serial line of the DLOG PC. The circuit diagram and printing drawing of the numerical display can be found on Fig.10.

6. Software architecture

Connection between the two PC-s of the Data Logging part is realised by WORKGROUPS FOR WINDOWS network (on ETHERNET cards), which gives possibility for both PCs to reach the collected data in real time and to provide control and set up possibility to each other.

The software architecture can be seen on Fig. ...... The software configuration of the DACQUS is based on 3 standard industrial software packages:
- LABTECH CONTROL I. for data logging,
- REALTIME VISION for real time representation,2

6.1. Terminology

The following list gives a short information about the most important naming used in technical description in connection with the LABTECH Control and VISION programs. The glossary description of the LABTECH terminology can be found in the LABTECH User's Guide.

Main features of the LABTECH software:
- menu-driven software
- icon-based programs

Setup

The setup means the fullness of the icon-based program assembled during the creation of the menu-driven software. In the Setup are involved from complete insertion of the Icons through the Icon connections to Network connections (see detailed in LABTECH User's Guide page No. 1).

Icon

The base (primary) component blocks of the LABTECH Control program are named Icons (that is an icon is a symbolic representation of signals, displays, logs, calculation blocks etc.). See detailed in LABTECH User's Guide page No. 11).

Analog and digital signal icons

They represents the input analog and digital signals

(signal connection to the hardware) with all signal parameters inclusive conversion parameters, alarm levels etc.

Log icons

A Log icon represent a disk file (with all logging and trigger parameters) used to store acquired data.

Icon view

ICONview provides a picture of application, showing the relationships and data flow between blocks (icons), and allows to use the icons as the actual building block of actual setup. With the aid of ICONview the user can define the settings for each block through dialogue boxes and the icons can be connected and moved with simple mouse commands. See detailed in LABTECH User's Guide page No. 11).

Build-Time

THE BuildTime of LABTECH allows to create a block diagram (called: setup).

Run-Time

The Run-Time means the running (execution) of the menu-driven and iconbased program in real time. The Run-Time executes the ICOview setup according to the settings specified in the Block and Log dialogue boxes. In addition, the Run-Time and VISION work together to display the setup data graphicly. The Run-Time (1) performs hardware data acquisition, block sampling and triggering; (2) passes data from block to block according to the connections in ICONview drawing, (3) creates and writes to logs, (3) displays a Blocks Run-Time window, (4) sends data to VISION displays (for graphical representation of collected data), (5) receives data from VISION display, and (6) creates DDE links to Windows applications. See detailed in LABTECH User's Guide page No. 259).

Few explanatory comments to the Log-Files

Log-file

Every log-file are identified with own name. The log-file characters can include the Date (@-sign) and version numbers of the similar log-files.

@-sign in Log name

Placeholder for the actual date. Form of the date is: ddmm (0304 means: 03 of April). With assistance of this option the Run-Time will create a new log-file at midnight (see detailed in LABTECH User's Guide page No. 158).

&-sign in Log name

Placeholder for the version numbers. The Run-Time will create a new log-file after every closing of the previous log-file. The version number in the name of the file will be incremented (see detailed in LABTECH User's Guide page No. 157).

Opening mode

This option determines how the Run-Time opens the log-file.

- Replace

  This setting causes the Run-Time to overwrite the data in the existing log-file (from the beginning of the log). Therefor, the log always contains the same number of records. This option in effect makes the log a "circular buffer" (see detailed in LABTECH User's Guide page No. 164).

- Append to Log

  This mode of the file opening causes the Run-Time to preserve the existing data log and start recording the new data at the end of the previous log. If the run time includes the header lines when it appends the log, so you can recognise where the new data begins (see detailed in LABTECH User's Guide page No. 164).

Closing mode

This form determines how the Run-Time closes a data log--file.

- On Time of Day

  This file closing mode causes the Run-Time to close the log-file and open a new one at a specified time interval. Wen you use this option, you must additionally enter an value for the Number of Hours. The log closing and opening occur on the hours or minutes of day, as determined by your computer's time of day clock. The interval is calculated from midnight. This option realises a synchronise to the time of day. For example if the Time is set 1, it means the log-file will close by 1 hours with

synchronised to the time of day (if the Time=24 the closing happen at midnight; if the Time=0,16667 the closing occur by 1 minutes with synchronised to the time of day; see detailed in LABTECH User's Guide page No. 165).

- After N Records

  This closing mode causes the Run-Time to close the log-file after the number of records (N) specified have been written to the log. At this option you must additional enter the value for N (see detailed in LABTECH User's Guide page No. 165).

Log on Condition

This entry is either checked (Yes) or unchecked (No). If it is No, the data log will be recorded by base sampling time (1 second) whenever the Start State is Enabled (Start/Stop methods). If it is yes, the data log will only be recorded when the selected block is in selected alarm condition (LABTECH User's Guide page No. 169). In our case the Start/Stop methods are controlled either Time of day controlled by Mod(X) (LABTECH User's Guide page No. 111) or inside CTRL signals (see: MANCTRL signal at MM@.prn log--file and SDCTRL signal at SD@.prn log-file). These controls determine either the time moment or time interval of the logging (LABTECH User's Guide page No. 49). The Sart/Stop control can be active either levels (On/Off) or edge (On Edge/Off Edge).

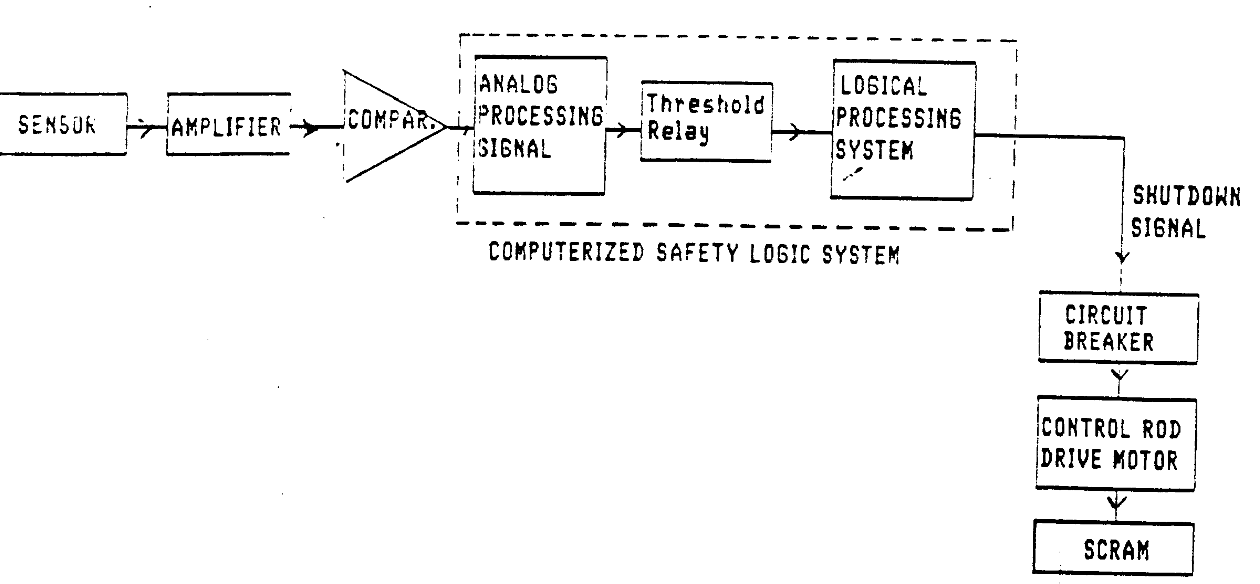
Pretransition Points

For Postmortem data logging we want to log some records (in our case 300 records) just before the reactor Shut-Down event occur (see detailed in LABTECH User's Guide page No. 165).

Samples in Logs

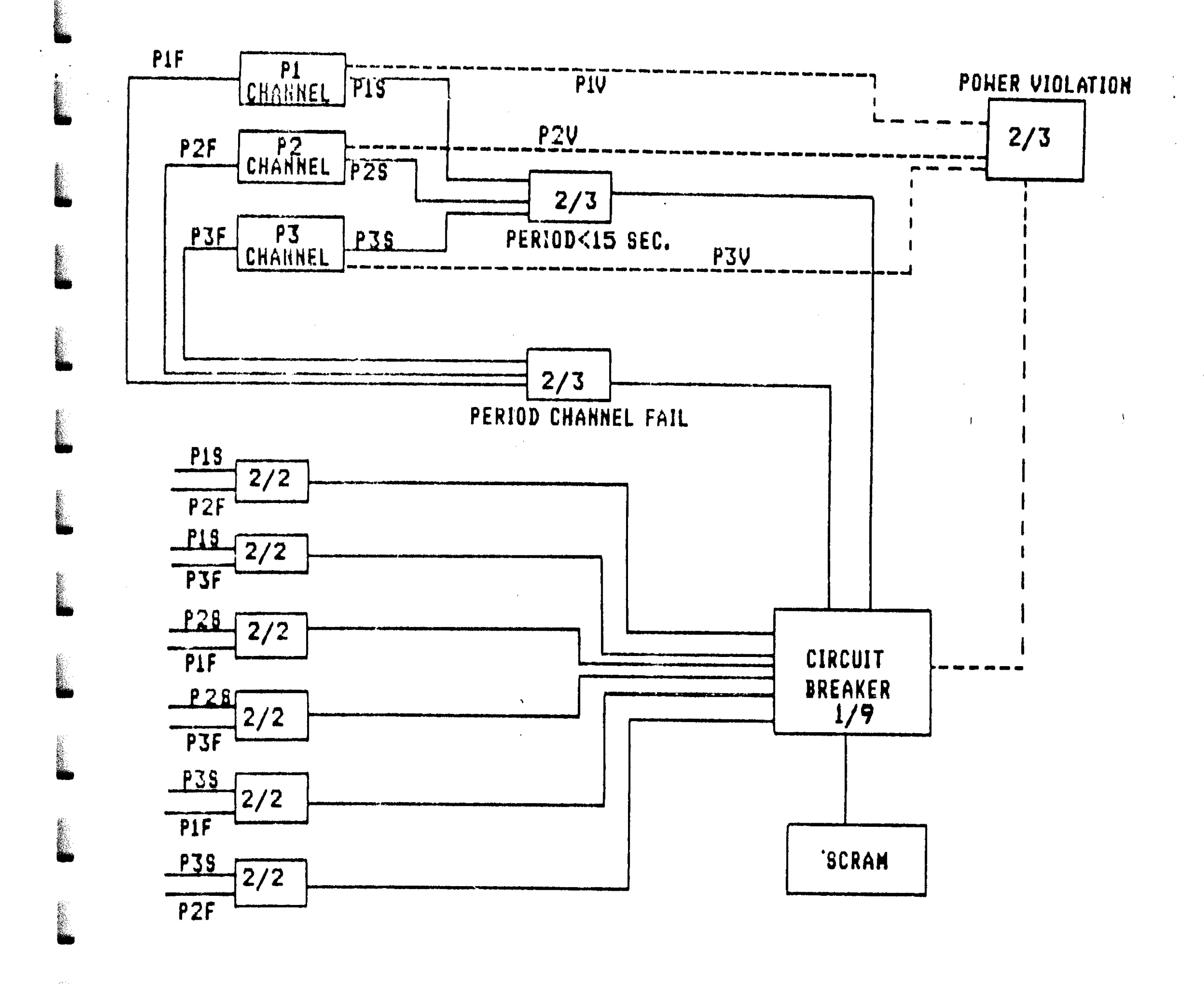
This number means the number of samples in logs by signals (row numbers of collected data).

* * * * *



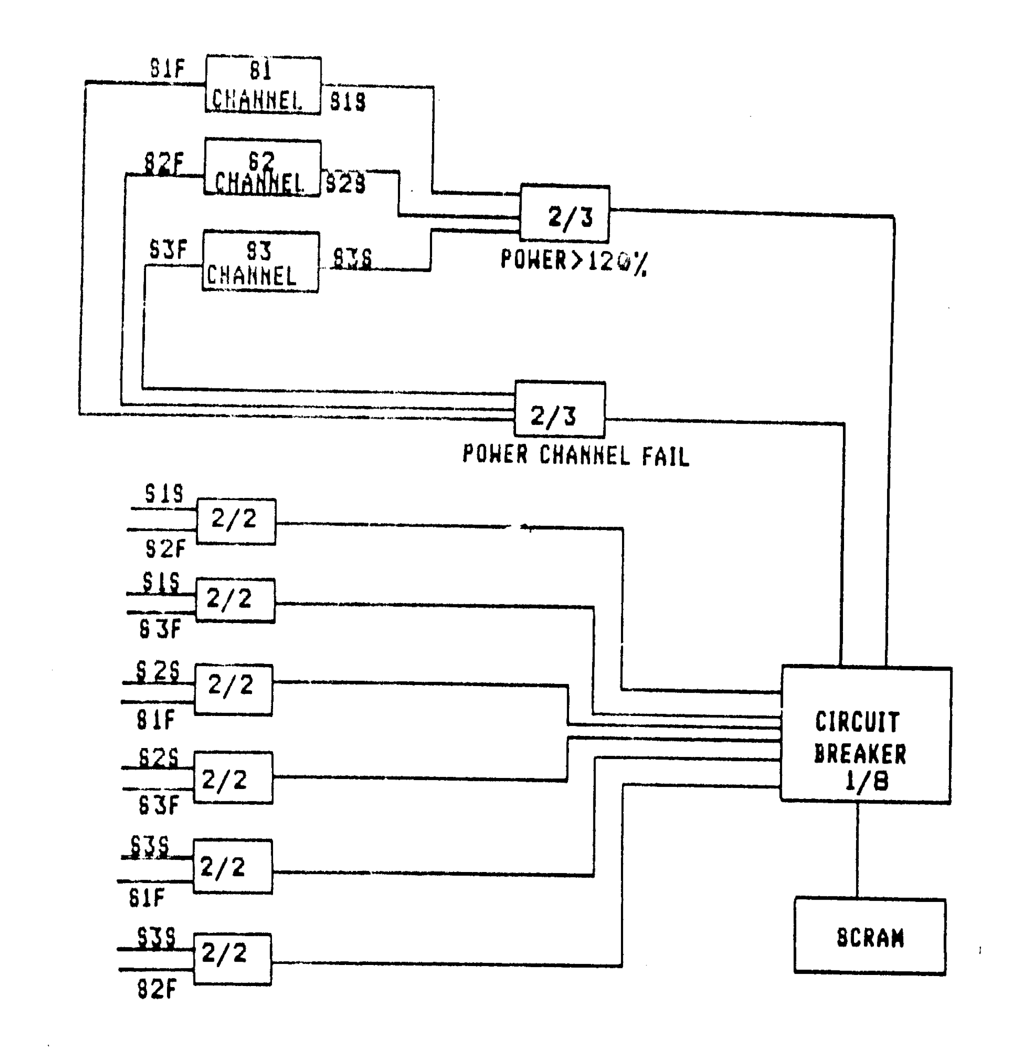
FLOW OF SCRAM SIGNAL IN ET-RR-1

Fig. 1

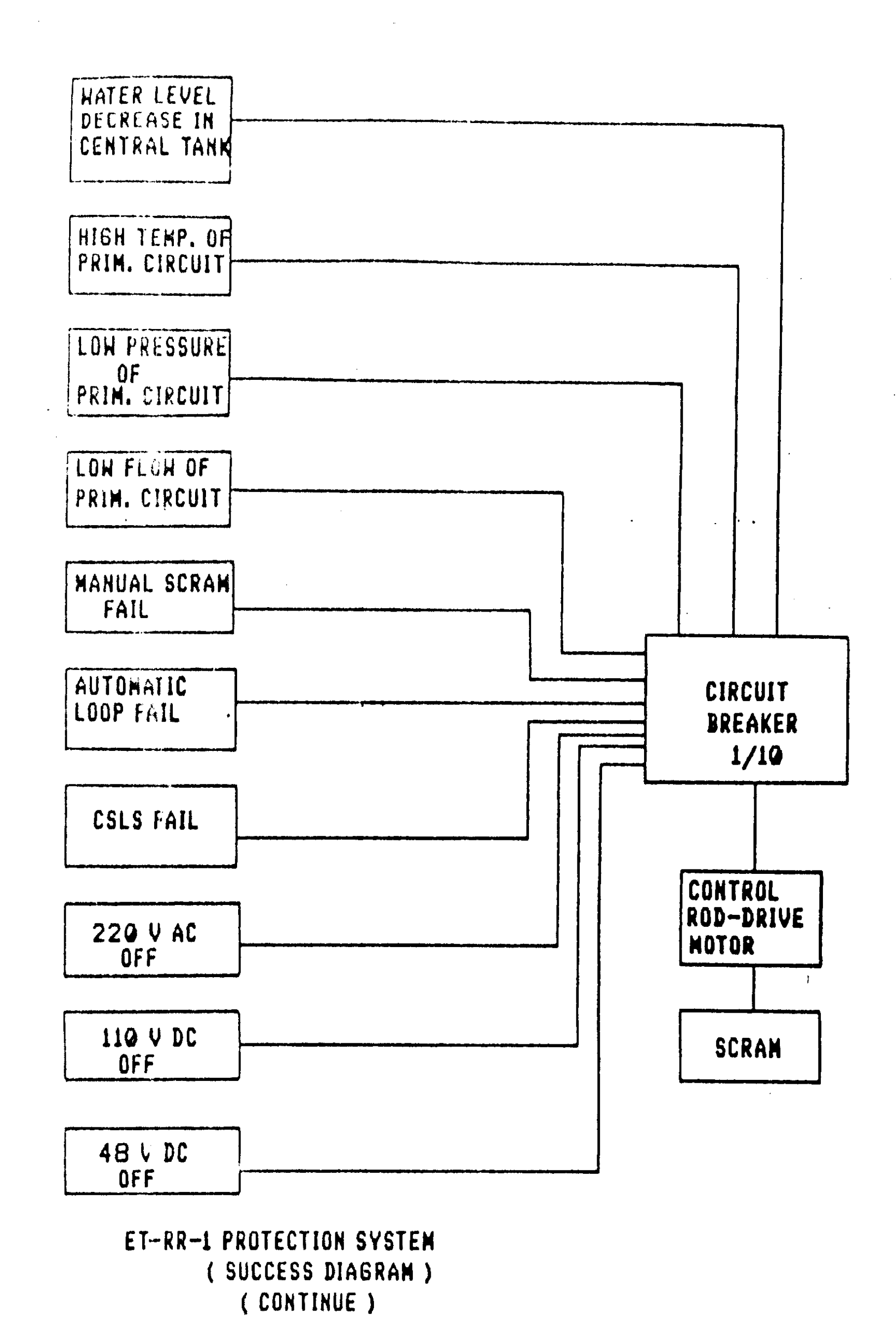


ET-RR-1 REACTOR PROTECTION SYSTEM
(SUCCESS DIAGRAM)

Fig.2



ET-RR-1 REACTOR PROTECTION SYSTEM
(SUCCESS DIAGRAM)
(CONTINUE)



ET-RR-1 PROTECTION SYSTEM
(SUCCESS DIAGRAM)
(CONTINUE)

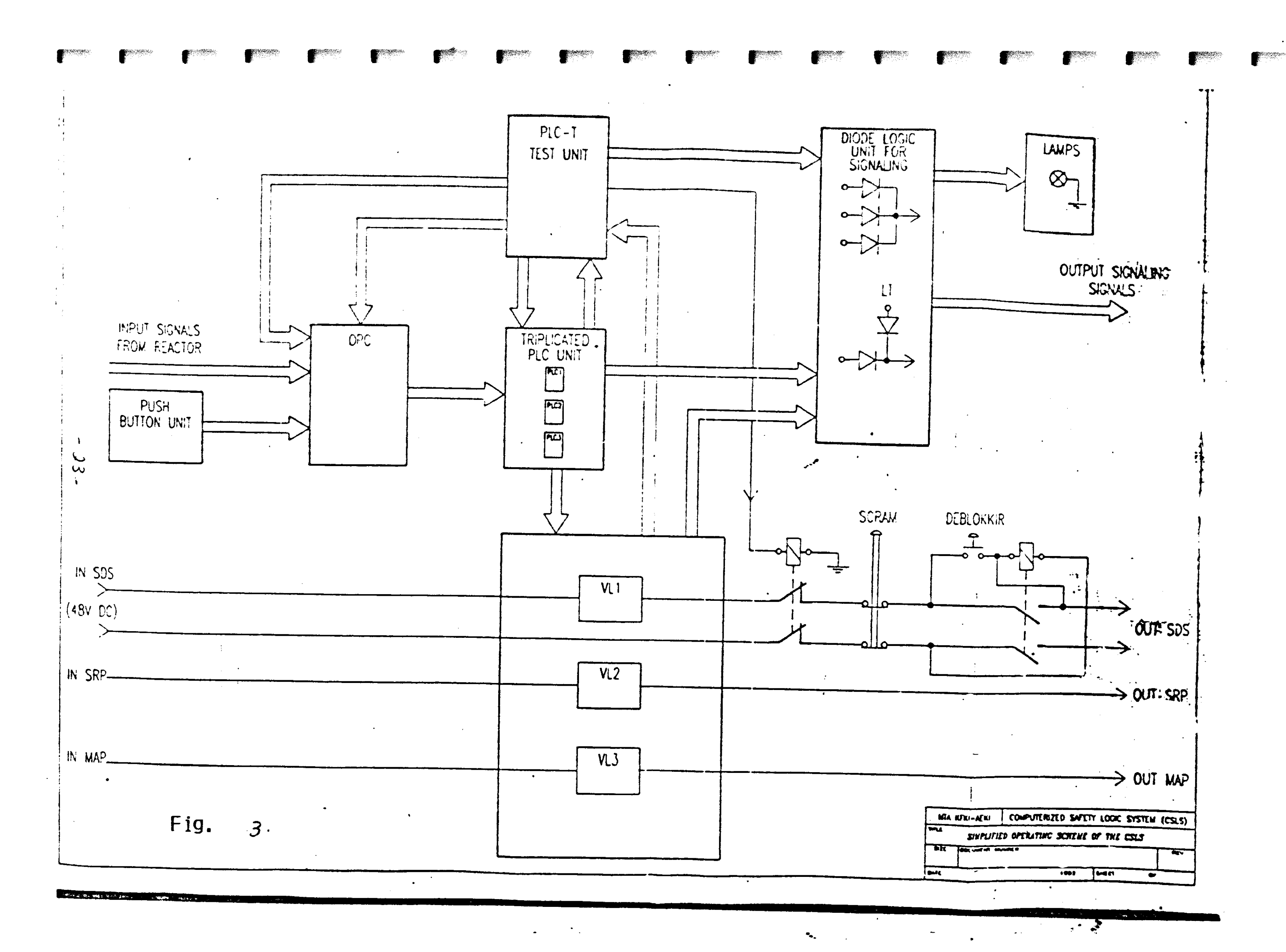

PLC-T
TEST UNIT
DIODE LOGIC
UNIT FOR
SIGNALING
LAMPS
OUTPUT SIGNALING
SIGNALS
LI
INPUT SIGNALS
FROM REACTOR
OPC
TRIPLICATED
PLC UNIT
PLC1
PLC2
PLC3
PUSH
BUTTON UNIT
SCRAM
DEBLOKKIR
IN SDS
(48V DC)
IN SRP
IN MAP
VL1
VL2
VL3
OUT SDS
OUT: SRP
OUT MAP
COMPUTERIZED SAFETY LOGIC SYSTEM (CSLS)
SIMPLIFIED OPERATING SCHEME OF THE CSLS

Fig. 3.